12 RADIATION PROTECTION

This chapter provides the results of the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff review of the United States - Advanced Pressurized Water Reactor (US-APWR) radiation protection as described in Chapter 12 of the US-APWR Design Control Document (DCD) submitted by Mitsubishi Heavy Industries, Ltd. (MHI), hereinafter referred to as the applicant, for the design certification (DC) of the US-APWR and the NRC staff referred to as the staff.

This chapter provides information on radiation protection methods and estimated occupational radiation exposure (ORE) of operating and construction personnel during normal operation and during anticipated operational occurrences (AOO). AOO may include refueling; purging; fuel handling and storage; radioactive material handling, processing, use, storage, and disposal; maintenance; routine operational surveillance; in-service inspection; and calibration. Specifically, DCD Chapter 12 provides information regarding facility and equipment design, planning and procedures, programs, techniques and practices employed by the applicant to meet the radiation protection standards set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 20, “Standards for Protection Against Radiation,”, and to be consistent with the guidance given in the appropriate regulatory guides used to implement the NRC regulations.

The NRC staff evaluated the information in DCD Chapter 12 using the criteria in Chapter 12 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition.” Compliance with these criteria provides assurance that radiation doses to workers will be maintained within the occupational dose limits of 10 CFR Part 20. These occupational dose limits, applicable to workers at the NRC licensed facilities, restrict the sum of the external whole body dose (deep-dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body (deposited through injection, absorption, ingestion, or inhalation) to 50 millisievert (mSv) (5 rem) per year with a provision (i.e., by planned special exposure) to extend this dose to 100 mSv (10 rem) per year with a lifetime dose limit of 250 mSv (25 rem) resulting from planned special exposures.

The Standard Review Plan (SRP) acceptance criteria also provide the guidance for assuring that radiation doses resulting from exposure to radioactive sources, both outside and inside the body, can be maintained well within the limits of 10 CFR Part 20 and as low as is reasonably achievable (ALARA). The balancing of internal and external exposure necessary to ensure that the sum of the doses is ALARA is an operational concern. An applicant seeking a combined license (COL) must address these operational concerns, as well as programmatic radiation protection concerns.

12.1 Ensuring that Occupational Radiation Exposures are As Low As Reasonably Achievable

12.1.1 Introduction

"As low as is reasonably achievable" (ALARA) means every reasonable effort is made to maintain exposures to radiation as far as practicable below the dose limits of 10 CFR Part 20. This includes taking into account the state of technology and the economics of improvements in relation to benefits to the public health and safety. It also includes using procedures and engineering controls based upon sound radiation protection principles.

12.1.2 Summary of Application

DCD Tier 1: DCD Tier 1 Section 2.8, "Radiation Protection," states that the US-APWR is designed to keep radiation exposures to plant personnel and members of the public within applicable regulatory limits and ALARA.

DCD Tier 2: The applicant has provided a Tier 2 design description Section 12.1, "Ensuring that Occupational Radiation Exposures are As Low As Reasonably Achievable," of the DCD, summarized here, in part, as follows:

The majority of nuclear plant worker ORE results from maintenance on systems that contain radioactive material, radioactive waste handling, in-service inspection, refueling, abnormal operations, and decommissioning work activities. These activities are addressed and included in the design of the US-APWR through the plant physical layout, selection of materials, shielding, and chemistry control.

During the design process the managers of specific engineering sections develop design specifications using the ALARA design requirements established by the manager responsible for radiation protection engineering. The applicant provides guidance to its staff regarding incorporating ALARA into the design including information on lessons learned from the nuclear power industry, from federal guidance, as well as almost forty years of industry research and analysis. An example of the use of lessons learned from operating plants is the use of zinc injection to reduce radioactive material build up and resultant exposure. Zinc is added to the primary coolant as an aqueous solution of zinc acetate, usually into the suction of the charging pumps or into the volume control tank. The zinc ions promote the formation of a more protective structured corrosion film on stainless steel. Since the corrosion film form favors zinc incorporation, available sites for ions have a higher probability of being filled with a zinc ion, than a cobalt ion. Therefore, the uptake of cobalt into the corrosion film will be significantly less if zinc ions are present in the water. For existing corrosion films, both zinc and cobalt compete for the same sites in the corrosion film. If the zinc concentration is high enough, the cobalt will be displaced from the existing corrosion film, thus reducing dose rates.

General design considerations and methods employed to maintain in-plant radiation exposures ALARA during operation, as well as during decommissioning, include minimizing the amount of personnel time spent in radiation areas, and minimizing radiation levels in routinely occupied plant areas near equipment expected to require personnel attention. Some of the features provided for maintaining exposures ALARA described in the US-APWR application, include:

- separation of less radioactive equipment from more radioactive equipment,
- provision of hatches for installation and removal of components,
- the use of Alloy 690 Steam Generator (SG) tubes to minimize cobalt introduction, and

- the use of Reactor Coolant System (RCS) chemistry controls that are optimized to minimize the production of corrosion products.

Inspection, Test and Analysis Acceptance Criteria (ITAAC): There are no ITAAC for this area of review.

Technical Specifications (TS): There are no TS for this area of review.

COL information or action items - (See Subsection 12.1.5 below).

Technical Report(s): There are no technical reports associated with this area of review.

Topical Report(s): There are no topical reports associated with this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues associated with this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements associated with this area of review.

Cross-cutting Requirements (Three Mile Island [TMI], Unresolved Safety Issue [USI]/Generic Safety Issue [GSI], Op Ex): There are no cross-cutting issues for this area of review.

10 CFR 20.1406 "Minimization of contamination": There is information pertinent to 10 CFR 20.1406 in Section 12.1.2.2.1, "General Design Criteria," and 12.1.2.3.2, "Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment."

12.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review and the associated acceptance criteria are given in Section 12.1 of NUREG-0800, the SRP, and are summarized below. No review interfaces with other SRP sections are listed in Section 12.1 of NUREG-0800.

Acceptance criteria are based on meeting the relevant requirements of the following NRC regulations:

1. 10 CFR Part 19 "Notices, Instructions and Reports to Workers: Inspection and Investigations," 19.12 "Instruction to workers," as it relates to keeping workers who receive ORE informed as to the storage, transfer, or use of radioactive materials or radiation in such areas, and instructed as to the risk associated with ORE, precautions and procedures to reduce exposures, and the purpose and function of protective devices employed.

2. 10 CFR 20.1101 "Radiation protection programs" and the definition of ALARA in 10 CFR 20.1003 "Definitions," as they relate to those measures that ensure that radiation exposures resulting from licensed activities are below specified limits and ALARA.

3. 10 CFR 52 “Licenses, Certifications, and Approvals for Nuclear Power Plants,” 52.47(b)(1), which requires that a DC application contains the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the COL, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

4. 10 CFR 20.1406 “Minimization of contamination,” which requires that applicants for DCs under Part 52 shall describe in the application how the facility design will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste.

Specific SRP acceptance criteria for the above requirements are as follows:

1. Policy Considerations: Acceptability will be based on evidence that a policy for ensuring that ORE will be ALARA has been formulated in accordance with the training requirements in 10 CFR 19.12 and the ALARA provisions of 10 CFR 20.1101(b), and that the policy has been described, displayed, and will be implemented in accordance with the provisions of Regulatory Guide (RG) 8.8 Revision 3, “Information Relevant for Ensuring that Occupational Radiation Exposures at Nuclear Power Stations is Reasonably Achievable” (Regulatory Position C.1), RG 8.10 Revision 1R, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable” (Regulatory Position C.1) and NUREG-1736, “Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation,” as it relates to maintaining doses ALARA. A specific individual will be designated and assigned responsibility and authority for implementing the ALARA policy. Alternative proposed policies will be evaluated on the basis of a comparison with the above regulatory guides and NUREG-1736.

2. Design Considerations: Acceptability will be based on evidence that the design methods, approach, and interactions are in accordance with the ALARA provisions of 10 CFR 20.1101(b) and RG 8.8 (Regulatory Position C.2) and will include: incorporation of measures for reducing the need for time spent in radiation areas; maintenance; measures to improve the accessibility to components requiring periodic maintenance or in-service inspection; measures to reduce the production, distribution, and retention of activated corrosion products throughout the primary system; measures for assuring that ORE during decommissioning will be ALARA; reviews of the design by competent radiation protection personnel; instructions to designers and engineers regarding ALARA design; experience from operating plants and past designs; and continuing facility design reviews. Alternative proposed design policies will be evaluated on the basis of a comparison with the design guidance in RG 8.8 (Regulatory Position C.2).

3. Operational Considerations: Acceptability will be based on evidence that the applicant has a program to develop plans and procedures in accordance with RG 1.33 Revision 2 “Quality Assurance Program Requirements (Operation)”, RG 1.8 Revision 3, “Qualification and Training of Personnel for Nuclear Power Plants”, RG 8.8, and RG 8.10 that can incorporate the experiences obtained from facility operation into facility and

equipment design and operations planning and that will implement specific exposure control techniques.

4. Radiation Protection Considerations: Acceptability will be based on evidence that the overall facility operations, as well as the radiation protection program, integrate the procedures necessary to ensure that radiation doses are ALARA, including work scheduling, work planning, design modifications, and radiological considerations.

12.1.4 Technical Evaluation

The staff compared the information in DCD Tier 2, Section 12.1, to the guidance in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Revision 1, as well as the criteria in Section 12.1 of the SRP regarding the radiation protection aspects of the reactor design. Specifically, the staff reviewed DCD Tier 2, Section 12.1 to ensure that the applicant had either provided information that was consistent with the guidance of the RGs and staff positions, referenced in Section 12.1 of the SRP, or had provided acceptable alternatives. As described below, the staff finds that DCD Tier 2, Section 12.1 conforms to the applicable guidance contained in these RGs and applicable staff positions. The staff also verified that there were no US-APWR Technical Reports, US-APWR Topical Reports or Westinghouse Commercial Atomic Power reports with information relevant to the review of DCD Section 12.1. The staff review of the information presented in DCD Section 12.1 did not identify a need for any non-editorial Requests for Additional Information (RAIs). The staff finds that the information provided in DCD Section 12.1, conforms to the guidance contained in the applicable RGs and staff positions, and therefore concludes that the relevant requirements of 10 CFR Part 20 have been met.

12.1.4.1 Policy Considerations

In DCD Tier 2, Section 12.1.1, "Policy Considerations," the applicant described the design, construction, and operational policies that have been implemented to ensure that ALARA considerations are factored into each stage of the design process. The applicant has committed to ensuring that the plant will be designed and constructed in a manner consistent with the guidance of RG 8.8. In particular, DCD Tier 2, Section 12.1.2, "Design Considerations," states that component designers and engineers have been instructed regarding ALARA design, and that procedures require the design engineer to consider the applicable RGs as part of the design criteria. The staff finds that the stated ALARA design and construction policy conforms to the guidance contained in RG 8.8 and is, therefore, acceptable.

12.1.4.2 Design Considerations

The plant radiation protection design should ensure that individual doses and total person-Sievert (person-rem) doses to plant workers and to members of the public are ALARA, and individual doses are maintained within the limits of 10 CFR Part 20. DCD Tier 2, Section 12.1.2 describes general design considerations and methods employed to maintain in-plant radiation exposures ALARA during operation, as well as during decommissioning, including minimizing the amount of personnel time spent in radiation areas and minimizing radiation levels in routinely occupied plant areas near plant equipment expected to require personnel attention for maintenance/surveillance. These design features address the following design considerations:

- Minimizing facility contamination to reduce radioactive waste generation, facilitate decommissioning, and to ensure that the facility can be operated and maintained with exposures ALARA.

- Reducing access, repair, and equipment removal times, thereby reducing the time spent in radiation fields.

- Reducing component radiation levels through material selection to reduce cobalt introduction rates, use of filtration systems to reduce corrosion product distribution, design of components and piping to avoid corrosion product collection points and selection of components to reduce or control equipment leaks.

- Minimizing personnel time spent in radiation areas by using facility layout and component locations to allow remote operation, inspections and calibrations, and by the provision, where practicable, of features to allow removal of components for transport to lower radiation areas for maintenance.

- Minimizing radiation levels through system design and facility layout and control of plant chemistry, thereby allowing operations, maintenance, and inspection activities to be performed in reduced radiation fields.

The US-APWR DCD Tier 2, Section 12.1.2 describes design features that ensure the plant can be operated, maintained and decommissioned with exposures ALARA, including:

- The design of systems include provisions for flushing and decontaminating equipment and piping, to remove accumulated radioactive material for dose rate reduction.

- Materials in contact with the RCS have low concentrations of cobalt, which reduces the amount of cobalt introduced into the RCS, and thus the amount of cobalt-60 that is produced. Cobalt-60 is the primary long-term source of radiation fields in nuclear power plants and, therefore, is a major source of radiation exposure during shutdown, maintenance, and inspection activities at light water reactors (LWR).

- Systems are segregated, such that radioactive systems are separated from nonradioactive systems, and radioactive components are located in separate shielded cubicles. This allows workers to perform maintenance on a given component, while being shielded from the dose of adjacent components.

- The use of zinc injection to reduce radioactive material build up and resultant personnel exposure.

- Ventilation systems are designed to minimize the spread of airborne contamination, by directing air flow from less contaminated areas, to more contaminated areas. To prevent exfiltration or infiltration of potential contaminants, the design ventilation exhaust flow rate is greater than supply flow rate in the areas with the potential for contamination.

- Equipment and piping are designed to minimize the accumulation of radioactive materials.

- Vents and drains from highly radioactive equipment are piped directly to the liquid waste collection system, preventing contaminated fluid from flowing across the floor to a drain and creating a potential airborne contamination problem.

- Flanged connections are provided for large vessels that may require decontamination prior to major maintenance activities. These connections are located outside the vessel room in order to shield the worker from the contents of the vessel, and minimize radiation exposure.

- Refueling tool surfaces are smooth to reduce contamination build-up on refueling equipment.

- Monitoring and control of primary-side water chemistry in accordance with the EPRI PWR Primary Water Chemistry Guidelines. Reactor coolant chemistry provides a constant pH in a target range that is optimized to minimize the production of corrosion products. By inhibiting primary-side corrosion-induced degradation through chemistry control the number of corrosion products that pass through the core is minimized and therefore the number of activated corrosion products that can contribute to worker exposure is reduced.

These design considerations incorporate the basic management philosophy guiding the design effort and they also conform to the guidance in RG 8.8. The design incorporates almost 40 years of research and analysis, and the lessons learned from the operating histories of 23 pressurized water reactor (PWR) power plants located in Japan. As a result, the design includes various improvements over prior designs. Examples include:

- An improved understanding of activated corrosion products behavior and buildup, which resulted in features, like the use of alloy 690 in the SGs, for minimizing activated corrosion products source generation and methods of reducing activated corrosion products buildup in normal operation, like the use of zinc addition to the RCS;

- A revised understanding of radiation streaming behavior, resulting in a better use of labyrinths and offsets for reduction of radiation streaming, and;

- The improvements in component design and configuration leading to a reduction of the time needed for maintenance and inspection.

The design features described in DCD Tier 2, Section 12.1.2 are intended to minimize personnel exposures and conform to the guidance contained in RG 8.8. As such, these design features will help to maintain individual doses and total person-rem doses to plant workers and to members of the public ALARA, while maintaining individual doses within the limits of 10 CFR Part 20.

12.1.4.3 Operational Considerations

The requirements of 10 CFR Part 20 specify that all licensees must develop, document, and implement a radiation protection program. Specifically, this program shall encompass the ALARA concept and include provisions for maintaining radiation doses and intakes of radioactive materials ALARA. The operational ALARA policy forms the basis for the operating station's ALARA manual. The applicant stated that the facility design, administrative programs and procedures ensure that occupational radiation exposure to personnel are maintained ALARA. The organization of responsibilities for the design and the operation of the US-APWR are intended to achieve ALARA occupational radiation exposures.

The DCD applicant stated, in DCD Tier 2, Section 12.1.3, "Operational Considerations," that in order to comply with the requirements of 10 CFR Part 20 and 10 CFR 19.12 and to maintain doses to plant personnel ALARA, the COL applicant is to provide the operational radiation protection program for ensuring that occupational radiation exposures are ALARA. Per DCD Tier 2, COL Information Item 12.1(5) this program is to be developed, implemented and maintained as described in the Nuclear Energy Institute (NEI) Technical Report, NEI 07-03A "Generic DCD Template Guidance for Radiation Protection Program Description", and NEI 07-08A "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)". The staff has reviewed NEI 07-03A (ML091490684) and NEI 07-08A (ML093220178) and finds them to be acceptable. DCD Tier 2, Section 12.1.3, "Operational Considerations," COL Information Item 12.1(1) and COL Information Item 12.1(3) state that the COL applicant will comply with the requirements of 10 CFR Part 20 and the guidance in the following regulatory guides:

- RG 1.8, Revision 3, "Qualification and Training of Personnel for Nuclear Power Plants"
- RG 8.2, Revision 0, "Guide for Administrative Practices in Radiation Monitoring"
- RG 8.4, Revision 0, "Direct-Reading and Indirect-Reading Pocket Dosimeters'
- RG 8.6, Revision 0, "Standard Test Procedure for Geiger-Muller Counters"
- RG 8.7, Revision 2, "Instructions for Recording and Reporting Occupational Radiation Exposure Data"
- RG 8.8, Revision 3, "Information Relevant for Ensuring that Occupational Radiation Exposures at Nuclear Power Stations is Reasonably Achievable"
- RG 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program"
- RG 8.10, Revision 1R, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable"
- RG 8.13, Revision 3, "Instruction Concerning Prenatal Radiation Exposure"
- RG 8.15, Revision 1, "Acceptable Programs for Respiratory Protection"

- RG 8.27, Revision 0, “Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants,”

- RG 8.28, Revision 0, “Audible Alarm Dosimeters”

- RG 8.29, Revision 1, “Instruction Concerning Risks from Occupational Radiation Exposure”

- RG 8.34, Revision 0, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses”

- RG 8.35, Revision 0, “Planned Special Exposures”

- RG 8.36, Revision 0, “Radiation Doses to Embryo/Fetus”

- RG 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants”

- The detailed policy considerations regarding overall plant operations and implementation of such a radiation protection program are outside the scope of the DC review. However, COL Information Items 12.1(1), 12.1(3), 12.1(5), 12.1(6) and 12.1(7) describe the program elements required to be provided by the COL applicant in order to implement the operational radiation protection and ALARA program requirements of 10 CFR Part 20.

12.1.5 Combined License Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the DCD:

Table 12-1
US-APWR Combined License Information Items

Item No.	Description	Section
12.1(1)	The COL applicant is to demonstrate that the policy considerations regarding plant operations are compliance with RG 1.8, RG 8.8 and RG 8.10 (Subsection 12.1.1.3).	12.1
12.1(2)	Deleted	12.1
12.1(3)	The COL applicant is to describe how the plant follows the guidance of RG 8.2, RG 8.4, RG 8.6, RG 8.7, RG 8.9, RG 8.13, RG 8.15, RG 8.20, RG 8.25, RG 8.26, RG 8.27, RG 8.28, RG 8.29, RG 8.32, RG 8.34, RG 8.35, RG 8.36 and RG 8.38.	12.1
12.1(4)	Deleted	12.1
12.1(5)	The COL applicant is to provide the operational radiation protection program for ensuring that occupational radiation exposures are ALARA.	12.1

Table 12-1
US-APWR Combined License Information Items

Item No.	Description	Section
12.1(6)	The COL applicant is to perform a periodic review of the operational practices to ensure configuration management, personnel training and qualification update, and procedure adherence.	12.1
12.1(7)	The COL applicant is to track implementation of requirements for record retention according to 10 CFR 50.75(g) and 10 CFR 70.25(g) as applicable.	12.1

12.1.6 Conclusions

Based on the information supplied by the applicant, as described above, the staff concludes that the US-APWR design policy and design features are acceptable. This conclusion is based on the applicant having met the ALARA provisions of 10 CFR 20.1101(b) and the applicable guidance of RG 8.8 (Regulatory Position C.2) and RG 8.10 (Regulatory Position C.1).

The objective of the plant radiation protection design is to maintain individual doses and total person-Sievert (person-rem) doses to plant workers, including construction workers, and to members of the general public, ALARA, and to maintain individual doses within the limits of 10 CFR Part 20. The ALARA design requirements are established by the manager responsible for radiation protection engineering. Component designers and engineers are given instructions regarding ALARA design methods, and design control procedures require the design engineer to consider the applicable RGs as part of the design criteria. This includes information regarding lessons learned from the nuclear power industry.

The applicant incorporated facility and equipment design considerations into the US-APWR design to reduce radiation exposure and to satisfy the radiation protection design objectives listed above. These included:

- Minimizing to the maximum extent possible, the cobalt content of alloys chosen for (RCS materials.
- Selecting materials with low corrosion rates.
- The use of modified pH control.
- Zinc injection, to reduce radiation exposure.
- Components are compartmentalized where necessary, and nonradioactive systems are separated from radioactive systems to minimize doses.

Doses during maintenance are reduced through the use of installed isolation, drain, and vent valves for draining, flushing or decontamination of systems. These design features, which are intended to maintain individual doses and total person-rem doses within the limits of 10 CFR Part 20, are in accordance with the guidance provided in RG 8.8 and are therefore acceptable.

Operating and maintenance personnel follow specific plans and procedures to ensure that goals related to keeping exposures ALARA are achieved in the operation of the plant. The ALARA operational implementation and policy considerations will be addressed by the COL applicant for the US-APWR. The material addressed by COL Information Items 12.1(1), 12.1(3), 12.1(5), 12.1(6) and 12.1(7) is site-specific and beyond the scope of the staff's review of the DCD. The staff will determine compliance with the requirements of 10 CFR Part 20 in these areas during the COL review.

12.2 Radiation Sources

12.2.1 Introduction

The determination of projected radiation sources during normal operations, AOO, and accident conditions in the plant, is used as the basis for designing the radiation protection program and for shield design calculations. This includes definition of isotopic composition, location in the plant, source strength and source geometry. In addition, the airborne radioactive material sources in the plant are considered in the design of the ventilation systems and are used for the design of personnel protective measures and for dose assessment.

12.2.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for the DCD Tier 2, Section 12.2 area of review.

DCD Tier 2: The applicant has provided a Tier 2 design description in Section 12.2 "Radiation Sources" of the DCD, summarized here, in part, as follows:

Section 12.2 discusses and identifies the sources of radiation that form the basis for the shielding design calculations and the sources of airborne radioactivity used to design personnel protection measures and perform dose assessments.

The shielding design-basis for primary coolant source term is based on US-APWR specific design inputs and a one percent failed fuel fraction. During normal operation, radiation within the containment consists of neutrons and gamma radiation emitted by the reactor core. Radiation levels are reduced by shielding provided by the reactor vessel and reactor internals.

The sources of radiation during normal full-power operation are direct core radiation, coolant activation processes, the leakage of fission products from defects in the fuel rod cladding, and the activation of the reactor coolant erosion and corrosion products. This radioactive material is continuously transported through the large reactor coolant piping. During operation, nitrogen-16 (N-16) which is formed by neutron interaction with oxygen in exposed water, is the largest source of radioactivity in the reactor coolant, reactor coolant pumps and SGs, and because of the associated 6.1 million electron volts (MeV) decay gamma, has the most impact on shielding design in the pre-stressed concrete containment vessel. While new N-16 is continuously produced as coolant passes through the reactor core, the N-16 activity in each of the primary coolant system components depends on the total transit time from the reactor core to the component and the residence time in the component. N-16 activity is not a factor in the radiation source term for systems and components located outside the containment due to its short half-life (7.35 seconds) and a transport time of greater than 1 minute before the primary coolant exits containment.

Fission and corrosion product activities circulating in the RCS and out-of-core corrosion products comprise the remaining significant radiation sources during full-power operation. During plant operation, radioactive non-gaseous fission and corrosion products deposit on the inner surface of pipes and components. This buildup of contamination is a continuous process, which is mainly dependent on physical and chemical conditions of the RCS in the different states of the reactor (full power, shutdown, and startup). The design basis of one percent failed operational fuel fraction is used as the source term to establish shielding provisions for the Auxiliary Building (AB). The fission and corrosion product activities circulating in the reactor coolant are given in US-APWR DCD Tier 2, Subsection 11.1.1. Subsection 11.1 estimates the design basis as well as the realistic source terms in the reactor coolant. In the shielding design, only the design basis reactor coolant source terms are considered. This source term is calculated using the ORIGEN code, but without using methods described in American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard (Std.) 18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors." Values of corrosion product concentrations are based on operating PWR reactor data.

The Chemical and Volume Control System (CVCS) processes reactor coolant for purification, degassing, and treatment. Some of the components of this system are located outside of containment. The design of the CVCS ensures that most of the N-16 decays before the letdown stream leaves the containment by using a long letdown flow path.

Airborne radioactivity concentrations in the Containment Building (CB), can occur as a result of coolant leakage (during power operation) and evaporation from the refueling pool (during refueling). The spent fuel pool (SFP) water contains radionuclides from defects in spent fuel and corrosion products from fuel assemblies. The evaporation of the SFP water can lead to airborne radioactivity concentrations in the Fuel Storage Building, both during power operation and refueling.

Airborne radioactivity concentrations within the AB and Reactor Building (RB) result principally from equipment leakage. The design of the ventilation systems in radiological portions of these buildings provides for airflow that is from regions of lower potential for contamination to those with higher potential for contamination. As a result, the airborne radioactivity concentrations are expected to be low in those areas of the buildings, which are normally occupied.

Under normal operating conditions (no leakage into the component cooling water system (CCWS)), components within the uncontrolled portions of the AB are not expected to contain radioactive material. Airborne radioactivity concentrations in the uncontrolled portions of the AB are therefore expected to be negligible. The ventilation from these areas is exhausted through the radiological portions of the plant ventilation system which is monitored by the plant vent exhaust radiation monitor.

Under normal operating conditions (no primary-to-secondary leaks and no leakage into the CCWS), components within the Turbine Building (TB) are not expected to contain high levels of radioactive material. Airborne radioactivity concentrations in the TB are therefore expected to be negligible. No radiation monitoring is provided for the TB ventilation exhaust system.

ITAAC: There are no ITAAC associated with the review of DCD Tier 2 Section 12.2.

TS: There are no TS for this area of review.

COL information or action items - (See Subsection 12.2.5 below).

Technical Report(s): There are no technical reports associated with this area of review.

Topical Report(s): There are no topical reports associated with this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues associated with this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements associated with this area of review.

Cross-cutting Requirements (Three Mile Island [TMI], Unresolved Safety Issue [USI]/Generic Safety Issue [GSI], Op Ex): TMI issues 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 50.34(f)(2)(vii) [NUREG-0737 "Clarification of TMI Action Plan Requirements," Task Action Plan Item II.B.2] and 10 CFR 50.34(f)(2)(xxvii) [NUREG-0737 III.D.3.3]. In addition, there is information pertinent to TMI action items in Section 1.9.3, Table 1.9.3-2,

12.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 12.2 of NUREG-0800, the SRP, and are summarized below. No review interfaces with other SRP sections are listed in Section 12.2 of NUREG-0800.

1. 10 CFR 20.1201, "Occupational dose limits for adults," 10 CFR 20.1202, "Compliance with requirements for summation of external and internal doses," and 10 CFR 20.1206, "Planned special exposures," as they relate to limiting occupational radiation doses.

2. 10 CFR 20.1203 "Determination of external dose from airborne radioactive material" and 10 CFR 20.1204 "Determination of internal exposure", as they relate to limiting average concentrations of airborne radioactive materials to protect individuals and control the intake (inhalation or absorption) of such materials.

3. 10 CFR 20.1207 "Occupational dose limits for minors," as it relates to limiting exposure to minors to one-tenth of limits for adults.

4. 10 CFR 20.1301 “Dose limits for individual members of the public,” as it relates to the determination of radiation levels and radioactive materials concentrations within the components of waste treatment systems.

5. 10 CFR 20.1801 “Security of stored material,” as it relates to securing licensed materials against unauthorized removal.

6. 10 CFR Part 50, GDC 61 “Fuel Storage and Handling and Radioactivity Control,” in Appendix A to 10 CFR Part 50, as it relates to systems that may contain radioactive materials.

7. 10 CFR 50.34(f)(2)(vii) and GDC 19 “Control Room,” as they relate to the acceptable radiation conditions in the plant under accident conditions, and the source term release assumptions used to estimate calculate those conditions.

8. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the COL, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

The SRP is not a substitute for the NRC’s regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the applicable NRC regulations.

The following RGs, standards, and NUREGs provide information, recommendations, and guidance and in general describe a basis acceptable to the staff for implementing the requirements of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1206, 10 CFR 20.1207, 10 CFR 20.1301, and 10 CFR 20.1801.

1. Regulatory Guide 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” as it relates to the assumptions used in evaluating the concentrations of radionuclides in containment and plant systems following a loss-of-coolant accident (LOCA).

2. Regulatory Guide 1.7, Revision 3, “Control of Combustible Gas Concentrations in Containment,” as it relates to methods for determining gaseous concentrations of radionuclides in containment following an accident.

3. Regulatory Guide 1.112, Revision 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” as it relates to complying with 10 CFR 20.1301 concerning the calculation of realistic radiation levels and radioactive materials source terms for the evaluation of waste treatment systems.

4. NUREG-0737, “Clarification of TMI Action Plan Requirements”, Task Action Plan Item II.B.2, as it relates to the identification of specific post-accident sources of radiation in the facility.

5. ANSI/ANS Std. 18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," as it relates to the establishment of typical long-term concentrations of principal radionuclides in fluid streams of light-water-cooled nuclear power plants.

12.2.4 Technical Evaluation

The staff reviewed the descriptions of the radiation sources given in DCD Tier 2, Section 12.2, "Radiation Sources," to assess conformance with the guidance in RG 1.206 and the criteria in Section 12.2 of the SRP. The applicant will use the contained source terms described in the DCD as the basis for the radiation shielding design calculations and personnel dose assessment. The applicant will use the airborne radioactive source terms in the DCD in the design of ventilation systems and for assessing personnel dose. The staff reviewed the source terms in the DCD to ensure that the applicant had either committed to follow the guidance of the RGs and staff positions set forth in Section 12.2 of the SRP, or provided acceptable alternatives, which are further described in the evaluations of the specific issues.

12.2.4.1 Contained Sources

In the DCD Tier 2, Section 12.2.1, "Contained Sources," the applicant describes the shielding design radiation source terms during normal full-power operation, shutdown, and design basis accident events. DCD Tier 2, Section 12.2.1 describes all large contained sources of radiation that are used as the basis for designing the radiation protection program and completing shield design calculations. These sources include the reactor core; the RCS; the CVCS; the liquid, gaseous, and solid radioactive waste systems; and other miscellaneous sources. For each of these contained sources, the applicant provided either the source strength by energy group or the associated maximum activity levels listed by isotope.

The sources of radiation during normal full-power operation are direct core radiation, coolant activation processes, the leakage of fission products from defects in the fuel rod cladding, and the activation of the reactor coolant corrosion and erosion products.

Direct radiation from the core is reduced by the design of the core internals and the primary shield wall. This results in dose rates less than 0.01 mSv/h (1 mrem/h) from radiation penetrating through the shield wall. The arrangement of the reactor coolant piping and shielding material results in streaming dose rates of less than 1 Sv/h (100 rem/h) at the reactor coolant pipe penetrations. Sources of radiation in the RCS are fission products released from fuel and activation and corrosion products that circulate in the reactor coolant. The design basis of one percent failed fuel fraction is used as the source term in the RCS to establish shielding provisions for the A/B. This is a methodology that exceeds the guidance of Section 12.2 of the SRP, which calls for a shielding source term based on 0.25 percent failed fuel. Because the assumed source term activity is higher, the shielding design is more robust. The more robust shielding design results in lower expected dose rates during normal operation and AOOs.

Section DCD 12.2.1.1.2 indicated that the activity values derived in DCD Section 11.1, used a core thermal power of 4,451 Mega Watt thermal (MWt). However, DCD Section 1.1.4 notes that in order to account for calorimetric error allowance 4540 MWt is used for core power level. As a result of this observation, the staff issued **RAI 128-1731, Question 12.02-2**, requesting that the applicant clarify the bases for the assumed power level. The applicant's response to this question dated January 21, 2009, stated that because the calculations used in Chapter 11 and Chapter 12 are not part of the safety evaluation, and NUREG-0017 Revision 1 "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," uses a core power of 100 percent, the use of 4,451 MWt was appropriate. Because the response conforms to the assumptions stated in NUREG-0017 and the guidance in the SRP, the staff finds the response acceptable. The staff confirmed that Revision 2 of the DCD, dated October 2009, contains the changes committed to in the RAI response. Therefore, the staff concludes that, **RAI 128-1731, Question 12.02-2,** is closed.

DCD Tier 2 Table 12.2-70 "Parameters and Assumptions for Calculating Spent Fuel Source Strength" states that fuel enrichment is 4.55 weight (wt) percent. However, DCD Tier 2 Table 4.2-1 "Fuel Assembly Design Specifications" lists fuel enrichment as five wt percent. In **RAI 427-2909, Question 12.02-17**, the staff asked the applicant to explain the discrepancy. The portion of the applicant's response to **RAI 427-2909, Question 12.02-17**, dated September 28, 2009, addressing this issue, stated that the value of five wt percent represents the maximum allowable enrichment for a single bundle, but due to core loading limitations, the average enrichment for a full core load would be less than 4 wt percent, therefore the value of 4.55 wt percent was considered to a bounding average value. Based on the applicant's response that the actual average core fuel enrichment is bounded by the analytical assumptions, the portion of **RAI 427-2909, Question 12.02-17**, related to spent fuel enrichment is resolved, and the portion of the question related to dose rates is discussed later in this section.

During operation, N-16, which is formed by neutron interaction with oxygen in exposed water, is the largest source of radioactivity in the reactor coolant system fluid, reactor coolant pumps and SGs, and the associated 6.1 MeV decay gamma has the most impact on shielding design inside containment. Since N-16 is continuously produced as coolant passes through the reactor core, the N-16 activity in each of the primary coolant system components depends on the total transit time to reach the component and the residence time in the component. N-16 activity is not a factor in the radiation source term for systems and components located outside the containment due to its short half-life, and a transport time of greater than 1 minute before the primary coolant exits containment.

Tritium is another coolant activation product present in the RCS. The primary sources of tritium in the RCS in a PWR are: (1) diffusion of tritium from the fuel through the zircaloy cladding; (2) neutron activation of boron in the burnable poison rods and subsequent tritium diffusion through the stainless steel cladding; and (3) neutron activation of boron, deuterium and ${}_{3}Li^{6}$ in the reactor coolant. The fission yield of tritium for Uranium-238 is approximately 0.01 percent. The guidance contained in NUREG-0800 "Standard Review Plan 12.2" notes under "Acceptance Criteria" that for PWRs designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations should be based on a primary coolant concentration of 3.5 μCi/g. Operational data from PWR plants that are using two year fuel cycles without recycling of RCS as Primary Make Up Water, indicates that RCS tritium activity is above one μCi/g during portions of the operating cycle. In DCD Table 11.1-9 "Realistic Source Terms" the value used for RCS tritium activity was listed as one μCi/g. For the purposes of occupational radiation exposure control, the value provided in Table 11.1-9 is not conservative with respect to the guidance provided in NUREG-0800, Section 12.2, and the observed operational data. Therefore, the staff issued **RAI 140-1732, Question 12.02-4,** asking the applicant to provide justification for the use of a non-conservative assumption. The applicant's response to this question, dated February 6, 2009, stated that the primary coolant tritium concentration used in the airborne activity calculations will be 3.5 µCi/g, which conforms to the guidance in the SRP, and is therefore acceptable. The staff confirmed that Revision 2 of the design control document (DCD) , dated October 2009, contained the changes committed to, in the RAI response. Therefore, **RAI 140-1732, Question 12.02-4,** is closed.

Corrosion product activities circulating in the RCS and out-of-core corrosion products comprise the remaining significant radiation sources during full-power operation. During plant operation, radioactive non-gaseous fission and corrosion products deposit on the inner surface of pipes and components. This buildup of contamination is a continuous process, which is mainly dependent on the physical and chemical conditions of the RCS in the different states of the reactor operation (full power, shutdown, and startup). The fission and corrosion product activities circulating in the reactor coolant are given in US-APWR DCD Tier 2, Subsection 11.1 which estimates the design basis as well as the realistic source terms in the reactor coolant.

In shielding design, only the design basis reactor coolant source terms are considered. As described by the applicant, this source term is calculated using the ORIGEN code, but without using methods described in ANSI/ANS Std. 18.1-1999, which provides a conservative basis for shielding design based on the assumption of operation with design basis fuel defects. Values of corrosion product concentrations are based on domestic operating PWR data.

Corrosion product source term in the US-APWR DCD Tier 2 Revision 1Table 11.1-9 “Realistic Source Terms” list RCS activity for cobalt (Co)-60 as 2.9E-04 (µCi/g) and for Co-58 as 2.5E-03 (µCi/g), while Table 12.2-9 “Isotopic Composition and Specific Activity of Typical Out-of-Core Corrosion Products in the primary coolant” lists RCS activity for Co-60 as 8.9E-04 (µCi/g) and Co-58 as 6.1 E-03 (µCi/g). In **RAI 128-1731, Question 12.02-1**, the applicant was asked to resolve this apparent discrepancy. The applicant’s response to **RAI 128-1731, Question 12.02-1,** dated January 21, 2009, stated that the concentration of corrosion products in the RCS shown in Table 12.2-9 are design-basis values, which are equivalent to the mean values of actual data obtained from U.S. plants as listed in Table 2-10 of NUREG-0017, Revision 1. The values in NUREG-0017, Revision 1 were obtained about thirty years ago, prior to the implementation of modern cobalt reduction programs and chemistry practices. Because the US-APWR uses low cobalt materials and RCS water quality control, the concentrations of Co-58 and Co-60 listed in DCD Tier 2 Revision 1 Table 12.2-9, are considered to be well on the conservative side. Based on the applicant’s response, **RAI 128-1731, Question 12.02-1,** is closed.

The CVCS processes reactor coolant for purification, degassing, and treatment. The regenerative, letdown, and excess letdown heat exchangers are located inside the CB. The radiation sources for these components include N-16 contained in the RCS fluid stream. Due to the long letdown flow path design of the CVCS most of the N-16 decays before the letdown stream exits containment. After RCS fluid leaves the CB, it passes through a number of components, including filters and demineralizers prior to storage in the CVCS Holdup Tanks (HUT), or return to the RCS. DCD Tier 2 Revision 1 Table 12.2-17 “Chemical and Volume Control System Radiation Sources Mixed Bed Demineralizer Activity (70 ft^3 of Resin)” listed some of the parameters and assumptions used to calculate source terms in plant components. However, the information presented in DCD Tier 2 Revision 1 Section 12.2, and the associated tables, did not provide all of the information needed to support the activity values listed in DCD Tier 2 Table 12.2-17 and similar tables. In order to support confirmatory calculations, the staff asked the applicant, in **RAI 168-1739, Question 12.02-14**, to provide all of the methods, models and assumptions required to determine the source term in plant components. The applicant's response to **RAI 168-1739, Question 12.02-14**, dated March 4, 2009, provided information, such as decontamination factors (DF) for filters and demineralizer beds that allowed the staff to perform confirmatory activity calculations. However, during the review of the response to this question, the staff noticed an inconsistency in the assumed density values used for the shielding calculations of some CVCS components. As a result, **RAI 168-1739, Question 12.02-14**, was closed, but the issue it raised remained open. The staff asked the applicant, in follow-up **RAI 427-2909, Question 12.02-22**, to correct the density values, and provide the basis for the assumed values. The applicant's response to **RAI 427-2909, Question 12.02-22**, dated September 28 2009, agreed to update the density values in Table 12.2-1 "Radiation Source Parameters", and provided the technical basis for the assumed values. However, the technical basis for the stated values was not incorporated into DCD Section 12.2. As a result, **RAI 427-2909, Question 12.02-22**, was closed, but the issue it raised remained open. Therefore, the staff issued **RAI 532-4019, Question 12.02-28**, requesting that the applicant revise DCD Tier 2 Section 12.2 of the DCD to include this technical basis and methodology information. The applicant's response to **RAI 532-4019, Question 12.02-28**, dated April 9 2010, agreed to update the information provided in DCD Table 12.2-1 "Radiation Source Parameters." Based on the applicant's response, **RAI 532-4019, Question 12.02-28**, was resolved. However, DCD Tier 2 Revision 2 Subsection 12.2 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 532-4019 Question 12.02-28** is identified as **Confirmatory Item 12.02-5** and the staff will confirm that this information is included in a future revision of the DCD.

In addition to being dependent on media efficiency, the amount of activity deposited in system demineralizers is also dependent on fluid stream flow rates. The same parameters that are used to determine the fluid activity concentrations that are used as part of the basis for the airborne activity concentrations, also impact the amount of activity deposited in filters and demineralizers. In **RAI 427-2909, Question 12.02-19**, the staff asked the applicant to describe some of the parameters used to determine stated airborne activity concentrations. The applicant's response to **RAI 427-2909, Question 12.02-19**, dated September 28, 2009, which addressed airborne activity concentrations, included CVCS system flow rate data that was used to develop the fluid activity concentrations that formed the basis of the stated airborne activity concentration. In this response, the fluid system flow rates assumed in the model for developing the activity results presented in DCD Table 12.2-72 "Reactor Cavity And SFP Water Specific Activity In Refueling /Shutdown (Except Tritium)", appeared to exceed the flow rate capabilities for some of the CVCS system purification components described in DCD Tier 2 Table 9.3.4-3, such as the Mixed Bed (MB) letdown demineralizer beds, the Regenerative Heat Exchanger, the Seal Water Heat Exchanger, and the Reactor Coolant Filters. Because these CVCS system flow rate parameters are also part of the assumptions used to determine the amount of activity deposited in plant filters and demineralizers, the staff issued **RAI 532-4019, Question 12.02-27**, asking the applicant to revise Section 12.2 of the DCD to use flow rate data consistent with the plant design basis described in DCD Section 9.3.4. The applicant's response to **RAI 532-4019, Question 12.02-27** dated April 9, 2010, acknowledged the inconsistent information and provided a revised CVCS purification flow rate. However, this revised CVCS system flow rate also exceeded the flow rate capacity of the relevant CVCS system components. Following a teleconference with the applicant on August 05, 2010, to discuss this inconsistent information, the applicant submitted a revised response to **RAI 532-4019, Question 12.02-27** dated September 14, 2010, proposing a change to DCD Subsection 9.3.4.1.2.3 which states that the CVCS system is capable of providing purification flow rates consistent with their assumed flow rate, when using the Residual Heat Removal System for letdown during shutdown. However, because the applicant's response does not address how this assumed flow is purified, this issue is not resolved. **RAI 532-4019, Question 12.02-27,** is associated with the above request and the staff identified this as **Open Item 12.02-1.**

The US-APWR design uses boric acid evaporators (BAE) to process RCS and reuse the concentrated boric acid and the BAE distillate output in the CVCS reactivity control system. The concentrated boric acid solution is stored in the boric acid tanks located in the AB and the distillate is stored in the Primary Makeup Water Tanks (PMWT). Because the concentration of non-gaseous activity in the RCS fluid that remains after passing through the process system filters and demineralizers is increased by the same factor that the boric acid concentration is increased, the BAE and tanks and components downstream of the BAE may contain activity in concentrations higher than that present in the RCS. Estimated source terms and the methodology used for calculating those source terms were not presented for the BAE, boric acid tanks (BAT) and PMWT. The applicant's response to **RAI 142-1733 Question 12.02-7** and **Question 12.02-8** dated February 6, 2009, provided dimensions and concentrations of the liquid and gas containing portions of the BAE and tanks. The activity in these components was based on increasing the concentration of boric acid from 200 ppm boron (ppmB) to 7,000 ppmB. The applicant described 200 ppmB as a shielding design value, which is determined in consideration of the amount of water discharged to the CVCS HUT (averaged over the operation period) and the dilution rate of the boron concentration. However, the staff pointed out that near the end of core life, reducing RCS boron concentration from 200 ppmB to about 50 ppmB, about the point where further reduction in RCS boron concentration would be done by the deborating demineralizers, would produce more water than the capacity of a CVCS HUT, so the concentration factor of 35 stated in the response would not provide a conservative value for shielding design. Therefore, the staff issued **RAI 532-4019, Question 12.02-30**, requesting that the applicant revise DCD Tier 2 Section 12.2 to use boric acid concentration factors that are consistent with the plant design basis. The applicant's response to **RAI 532-4019, Question 12.02-30,** dated April 9, 2010, stated that the possibility of an increase in radioactive concentration can be seen only during the end of a cycle when the boron concentration is low and the concentration rate in the BAE is high. The applicant added COL Information Item 12.3(9) to confirm the radiation levels in the BAE rooms during the end of cycle instead of changing the source strength by increasing the concentration rate of the BAE. The purpose of defining the amount of activity contained in the BAE is to ensure that adequate shielding is provided for personnel protection. The activity contained within the BAE is based on the amount of concentration needed to raise boric acid concentration to the desired 7,000 ppmB, but it is also based on the inlet radioactivity concentration. The BAE are provided as a method of improving operating efficiency, and are not required to support plant operation. Based on the use of controls and monitoring for limiting BAE operation to ensure that the ORE remains ALARA, the applicant's method of describing the source term of the BAE conforms the guidance in the SRP, therefore the staff finds the applicant's response acceptable. Therefore, **RAI 532-4019, Question 12.02-30,** was resolved. However, DCD Tier 2 Revision 2 Subsection 12.3 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 532-4019 Question 12.02-30** is identified as **Confirmatory Item 12.02-6**,and the staff will confirm that this information is included in a future revision of the DCD.

The physical processes that the BAE uses to increase the concentration of boric acid in the RCS fluid stream causes deposits of radioactive material on internal surfaces of the BAE package that will remain even after the equipment is drained for maintenance. Staff operating experience, supplemented by documented industry operating information, indicates that this internal accumulation of material may impact the ability to work on components like pumps and instrument sensors that are located near major portions of the BAE package. Due to the high dose rates and the potential for high airborne activity and resultant internal exposure from handling this residual activity, in **RAI 142-1733, Question 12.02-8, and RAI 142-1733, Question 12.02-9**, the applicant was asked to describe the shut down external gamma radiation and airborne activity source terms for plant personnel working in the BAE vault. The applicant's response to **RAI 142-1733, Question 12.02-8,** and **Question 12.02-9** dated February 6, 2009, indicated that because the design employed multiple filters and demineralizers in the feed stream to the BAE, that insignificant amounts of radioactive material would be present in the BAE. Because this information appeared to be inconsistent with the NRC staff experience and some industry literature about ORE from BAEs at some operating plants with coolant activity levels well below the design basis activity concentrations specified for the US-APWR, **RAI 142-1733, Question 12.02-8, and RAI 142-1733, Question 12.02-9**, are closed, but the issues they addressed regarding source terms within BAEs remained open. As a result, in **RAI 427-2909, Question 12.02-18,** the applicant was asked to provide additional information about the source terms expected within the BAEs that were supported by the previously stated DF for filters and demineralizers. The applicant's response to **RAI 427-2909, Question 12.02-18,** dated September 28, 2009, provided source terms for the BAE in Table 12.2-66 "Chemical and Volume Control System Radiation Sources B.A. Evaporator Activity" that are based on the design basis failed fuel fraction**.** The applicant stated that because of the leak tight design of the BAE, that airborne activity around the BAE during operation would be minimal. The applicant also provided the BAE Vent Condenser Source Strength in Table 12.2-69 "Chemical and Volume Control System Radiation Sources B.A. Evaporator Vent Condenser Source Strength" that are based on the design basis failed fuel fraction. The NRC staff agrees with the applicant's statements regarding dose rates outside of the BAE vault, and the airborne activity concentrations inside the BAE vault during operation. However, the staff concluded that in its response to **RAI 427-2909, Question 12.02-18,** the applicant provided insufficient information regarding the estimated personnel exposure and the associated basis, for routine maintenance and surveillance of the BAE packages. **RAI 427-2909, Question 12.02-18,** is closed, but the issue it addressed remained open. Therefore, the staff issued **RAI 532-4019, Question 12.02-24**, and **RAI 532-4019, Question 12.02-30**, requesting that the applicant revise DCD Tier 2 Section 12.2 and Section 12.4 to describe the estimated personnel exposure and the associated source term basis, for routine maintenance and surveillance of the BAE packages. The applicant's response to **RAI 532-4019, Question 12.02-24**, dated April 9, 2010, described the applicant's operating equipment experience and current radiation protection program practices that formed the basis for the applicant's assertion that personnel exposure would be ALARA and was adequately described by information already contained in the DCD. The use of relevant experience from operating reactors to establish the basis for source terms conforms to the guidance contained in the SRP, so based on the applicant's response, **RAI 532-4019, Question 12.02-24,** is closed. The applicant's response to **RAI 532-4019, Question 12.02-30**, dated April 9, 2010, stated that the possibility of an increase in radioactive concentration can be seen only during the end of a cycle when the boron concentration is low and the concentration rate in the BAE is high. The applicant committed to adding COL Information Item 12.3(9) to confirm the radiation levels in the BAE rooms during the end of cycle instead of changing the source strength by increasing the concentration rate of the BAE. Based on the applicant's response, **RAI 532-4019, Question 12.02-30** was resolved. However, DCD Tier 2 Revision 2 Subsection 12.3 has not yet been updated to include the information provided in the response to the above question. **Therefore, RAI 532-4019, Question 12.02-30** is identified as

Confirmatory Item 12.02-6 and the staff will confirm that this information is included in a future revision of the DCD. DCD Tier 2 Revision 1 Section 12.2.1 provided a brief description of the Refueling Water Storage Auxiliary Tank (RWSAT), and the PMWT, which are located outside of the nuclear island block (AB, RB and CB). The RWSAT stores a portion of the water used to support refueling activities. The two PMWT receive the distillate output of the BAE. In DCD Tier 2 Revision 1 Chapter 12, the applicant stated that these three tanks would be located in concrete shielded enclosures. However, the applicant did not provide information regarding the configuration of the shielding and the resultant radiation zone near these tanks. In **RAI 144-1738, Question 12.02-12**, the staff asked the applicant to provide this information. The applicant's response to **RAI 144-1738, Question 12.02-12**, dated February 6, 2009, stated that they had changed the design to remove the concrete shielding surrounding these tanks. The applicant stated that exposure controls for workers and members of the public would be met by controlling the activity of the water in the tanks, and by limiting access to the area near the tanks. However, the applicant did not provide a COL information item to identify the need to limit the activity concentration in these tanks. The applicant did not reflect the use of barriers, and the resultant radiation zone for the areas surrounding these tanks on DCD Tier 2 Figure 12.3-1 "Radiation Zones for Normal Operation/Shutdown Site (Sheet 1 of 34)". The response also did not address the requirements of 10 CFR 20.1406 "Minimization of contamination," with respect to minimizing contamination of the environment. **RAI 144-1738, Question 12.02-12**, was closed but the issue it raised remained open. In follow-up **RAI 427-2909, Question 12.02-21**, the staff asked the applicant for additional information regarding how these tanks would meet regulatory requirements and the applicable regulatory guidance. The applicant's response to **RAI 427-2909, Question 12.02-21**, dated September 28, 2009, stated that the tanks would be located within a tank house, and the applicant committed to changing DCD Figure 12.3-1 to include the radiation zone information for the area around these tanks. This information is not included in DCD, Tier 2, Revision 2, Section 12.2.1.1.10, in Section 12.2.3 "Combined License Information" or in Figure 12.3-1. This portion of the response to **RAI 427-2909, Question 12.02-21,** is being tracked as **Confirmatory Item 12.02-1** and the staff will confirm that this information is included in a future revision of the DCD. The response to **RAI 427-2909, Question 12.02-21,** indicated that controls limiting the activity in the tanks would be part of some program, such as the Radiation Protection Program, and that dose rate at 2 meters from the tank surface will be limited to less than 0.25 mrem/h. However, the "Impact on the DCD" section of the response did not provide a COL information item stating the requirement to provide a program to limit tank activity. Additionally, because the applicant provided insufficient information regarding the design of the tank area enclosure, the NRC staff is unable to evaluate the adequacy of design features of the facility, such as the entry and egress points for the area, and the ventilation controls and effluent monitoring for the area. Therefore the staff issued **RAI 532-4019, Question 12.02-29** asking the applicant fully and accurately describe the tank enclosure facility and the associated required COL information items, **RAI 532-4019, Question 12.02-29,** is associated with the above request and the staff identified this as **Open Item 12.02-2.**

The core activity release model for a core melt accident is based on the source term model from RG 1.183. The applicant used the resulting source strengths to calculate post-accident dose rates, as well as worker doses incurred during vital area access/activities following an accident. In the event of core degradation, core cooling can be provided by four separate safety injection trains each located in a separate vault in the RB adjacent to the CB. The safety injection pumps take suction from the Refueling Water Storage Pit (RWSP) which is located inside the containment, thereby eliminating the need for switching Emergency Core Cooling System water supply to an external water source. The radioactivity in the recirculation water is calculated based on the assumptions that all the radioactive material released into the containment, except for noble gas, is dissolved in the fluid. Decreases due to deposit and attachment or being airborne in the containment are not taken into consideration. The Residual Heat Removal System (RHRS) and shielding are designed to allow limited access to the RHRS pumps following a Design Basis Accident (DBA). The staff confirmed the applicant's use of the RG 1.183 source term model and therefore their compliance with the requirements of 10 CFR 50.34(f)(2)(vii) and NUREG-0737 (II.B.2). Therefore, the staff finds the use of this accident source term acceptable.

The guidance in Section 12.2 of the SRP is that the DCD will provide sufficient information to allow the reviewer to determine whether source strengths, concentrations of airborne radioactivity, and quantitative source descriptions are consistent with the assumptions made and the methods used by the applicant. However, the information provided in DCD Tier 2 Section 12.2.1 regarding the In Core Instrument System (ICIS) only provided source strength data, without the supporting methods and assumptions used to derive the values. In **RAI 141-1735, Question 12.02-6**, the staff asked the applicant to provide the supporting methods and assumptions used to derive the values. The applicant's response to **RAI 141-1735, Question 12.02-6**, dated February 6, 2009, provided the requested information for the detector drive cable. However, as noted in the DCD Tier 2 Section 7.7.1.5.2, the ICIS detectors are miniature fission detectors. DCD Table 12.2-71 "Parameters and assumptions for Calculating Irradiated Incore Detector, Drive Cable and Flux Thimble Source Strength" does not show the source term and dose rates associated with these fission detectors. Therefore, **RAI 141-1735, Question 12.02-6,** is considered closed, however, the issued it raised remained open. In follow up question **RAI 427-2909, Question 12.02-17**, the applicant was asked to describe the dose rates, and their bases, for the fission chambers of the ICIS neutron detectors. The applicant's response to **RAI 427-2909, Question 12.02-17,** dated September 28, 2009, provided information related to the neutron detecting fission chambers attached to the drive cables. Because the applicant did not state that the provided information would be included in the DCD, **RAI 427-2909, Question 12.02-17,** is considered closed but the issue it raised remained open. The staff issued follow-up **RAI 532-4019, Question 12.02-23**, requesting that the applicant include the source strength calculational methods in DCD Tier 2 Section 12.2.1. The applicant's response to **RAI 532-4019, Question 12.02-23**, dated April 9, 2010, updated the information in Table 12.2-71 "Parameters and Assumptions for Calculating Irradiated Incore Detector, Drive Cable and Flux Thimble Source Strength", and Subsection 12.2.1.2.5 to include additional information regarding the ICIS detectors. Because the information provided regarding the incore instrument system detectors conforms to the guidance contained in the SRP, the staff concludes that, **RAI 532-4019, Question 12.02-23,** was resolved. However, DCD Tier 2 Revision 2 Subsection 12.2 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 532-4019, Question 12.02-23,** is identified as **Confirmatory Item 12.02-2** and the staff will confirm that this information is included in a future revision of the DCD.

The applicant's response to **RAI 427-2909, Question 12.02-17,** dated September 28, 2009, provided insufficient information about neutron detecting fission chambers attached to the drive cables. Therefore, the staff also issued **RAI 561-4441, Question 12.02-31,** asking the applicant to include the information regarding the uranium content of the ICIS detector fission chambers, and resultant isotopic concentrations in DCD Tier 2 Section 12.2.1. The applicant's response to **RAI 561-4441, Question 12.02-31**, dated April 9, 2010, referred to the applicant's response to

RAI 532-4019, Question 12.02-23, dated April 9, 2010, which committed to updating the information in Table 12.2-71 “Parameters and Assumptions for Calculating Irradiated Incore Detector, Drive Cable and Flux Thimble Source Strength,” and Subsection 12.2.1.2.5 to include additional information about the ICIS detectors. Because the information provided about the ICIS detectors conforms to the guidance contained in the SRP the staff concludes that, **RAI 561-4441, Question 12.02-31**, was resolved. However, DCD Tier 2 Revision 2 Subsection 12.2 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 561-4441, Question 12.02-31,** is identified as **Confirmatory Item 12.02-7** and the staff will confirm that this information is included in a future revision of the DCD.

The guidance in Section 12.2 of the SRP states that the applicant should include descriptions of any radiation sources containing byproduct, source, and special nuclear materials. The information provided by the DCD applicant conforms to the guidance in the SRP for describing DCD required sources that may require related facility design features. In DCD Tier 2, Section 12.2.1.1.10, “Miscellaneous Sources,” the applicant stated that the COL applicant will address any additional contained radiation sources that are not already identified in DCD Section 12.2.1. The DCD applicant identified this issue as COL Information Item 12.2(1). Because the DCD contains the appropriate COL action item for requiring the COL applicant to provide information about site-specific contained sources, that are outside of the scope of the DC, which will be provided by the COL applicant, the DCD conforms to the guidance in the SRP.

Operating PWR data for normal operation shows that due to technological improvements in nuclear fuel integrity resulting in a reduced fuel defect fraction, fission products in the reactor coolant in currently operating plants are normally negligible. In the shielding design, the activity of fission and corrosion products in the SFP water is conservatively estimated as exclusively Co-60. The SFP cooling pump suction line is connected to the SFP at an elevation of approximately four feet below the normal SFP water level. The return line contains a siphon breaker located near the surface of the water. These features are provided so that the pit cannot be gravity drained below a point of approximately 24 feet above the top of the spent fuel assemblies, thus maintaining the minimum SFP water level for radiation shielding of 11 feet 1 inches for a raised fuel assembly. Confirmatory calculations performed by the staff show that the dose rate at the water surface due to the combined radiation from a raised spent fuel assembly during fuel handling and the contaminated water in the SFP is 0.15 mSv/h (15 mrem/h), at normal SFP water levels, and is only fractionally higher at the minimum water level for radiation shielding of 11 feet 1 inches above the top of the fuel.

For the purpose of evaluating the design of SG blowdown system, the radioactivity in the main steam system is based on a SG tube leakage rate of 150 gallons per day concurrent with a one percent failed fuel fraction. Continuous operation with primary-to-secondary leakage is assumed. The RCS radionuclide concentrations used are those tabulated in DCD Tier 2 Revision 2, Table 11.1-2. This radioactivity is sufficiently low that no permanent radiation shielding is normally needed for equipment in secondary systems, other than portions of the SG blowdown system. The source terms for the SG blowdown demineralizer are tabulated in DCD Table 12.2-35 “SG Blowdown Demineralizer Activity (350 ft3 of Resin)”. Those portions of the SG blowdown system that are may require shielding to maintain radiation zone requirements, such as the SG blowdown system demineralizers, are contained within shielded cubicles located within the AB. As described in DCD Tier 2, Revision 2, Section 10.4.6, “Condensate Polishing System”, in the event of radioactive contamination of the resin in a condensate polisher media vessel, temporary shielding is installed if required. As such, these design features of the SG blowdown system will help to maintain individual doses and total person-rem doses to plant workers and to members of the public ALARA, while maintaining individual doses within the limits of 10 CFR Part 20. The description of the contained source terms and shielding provisions of the SG blowdown system conforms to the guidance provided in the SRP.

The US-APWR DCD states that shielded space is available within the plant for the storage of spent filters and resins. Any additional facilities for the storage of radioactive waste are the responsibility of the COL applicant. COL Information Item 12.2(2) states that the COL applicant is to describe any additional site specific radioactive waste storage facilities. Because design and source term information for these facilities are not within the scope of the DCD review, this conforms to the guidance contained in the SRP for identification of radiation sources, and the staff finds COL Information Item 12.2(2) acceptable.

12.2.4.2 Airborne Radioactive Material Sources

The guidance contained in RG 1.206 states that the applicant should describe in Section 12.2 of the DCD Tier 2, those airborne radioactive sources in the plant that are considered when designing the ventilation systems and in specifying appropriate monitoring systems. This description should include a tabulation of the calculated concentrations of radioactive material, by nuclide, expected during normal operation, AOO, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. It should also include models and parameters for the calculations. In DCD Tier 2, Section 12.2.2, "Airborne Radioactive Material Sources," the applicant described the sources of airborne radioactivity for the reactor design. The applicant stated that radioactive material that becomes airborne may come from the RCS, spent fuel pit, and RWSP. The calculation of potential airborne radioactivity in equipment cubicles, corridors, or operating areas normally occupied by operating personnel is based on reactor coolant activities given in DCD Tier 2, Chapter 11, Section 11.1. While DCD Tier 2 Table 12.2-60 "Parameters and Assumptions for Calculating Airborne Radioactive Concentrations" contained some information about the parameters used to calculate airborne activity concentrations, insufficient information was provided to allow the staff to confirm the airborne activity concentrations presented by the applicant. The staff issued **RAI 143-1737, Question 12.02-10**, to request this information. The applicant's response to **RAI 143-1737, Question 12.02-10**, dated February 6, 2009, provided the parameters and assumptions for calculating airborne activity levels. However, the information provided by the applicant did not include enough information regarding ventilation supply and removal rates and room volumes to allow the determination of removal rates. The applicant's response used an assumed leakage rate that is much less than the allowable TS leakage rate. The Purge Flow Duration for Low Volume Purge assumption was listed as "Continuous," but Chapter 16 TS Section 3.6.3.2 indicates that the containment isolation valves are normally shut. Because there was insufficient information available to allow the NRC staff to confirm the airborne activity concentration values provided in DCD Table 12.2-61 "Airborne Radioactive Concentrations", **RAI 143-1737, Question 12.02-10**, was considered closed but the issue it raised remained open and the staff issued follow-up **RAI 427-2909, Question 12.02-19**, asking the applicant to clarify and the correct the information already provided. The applicant's response to **RAI 427-2909, Question 12.02-19**, dated September 28, 2009, provided additional parameters and source term information for use in calculating airborne concentrations, including the simplifying assumption of ignoring radioactive decay when calculating the equilibrium airborne radioactivity concentration. This simplifying assumption essentially reduces the airborne radioactivity model to a ratio of the constant leakage into the area to the constant ventilation removal rate. The information provided by the applicant was enough to allow the staff to perform airborne activity concentrations that confirmed the data provided by the applicant. However, in its response to **RAI 427-2909, Question 12.02-19**, the applicant asserted that the US-APWR DCD Tier 2, Chapter 16 TS Surveillance Requirement 3.6.3.2 allows an exception for the normally closed valve position of the Low Volume Purge containment isolation valves, for the purpose of air quality considerations, so the use of Low Volume Purge on a continuous basis for reducing airborne activity concentrations in the CB was acceptable. The position of the NRC staff responsible for containment isolation is that this answer is not acceptable. The staff position is supported by Branch Technical Position (BTP) 6-4 Revision 3 "Containment Purging During Normal Plant Operations" Section B.3 which states "The need for purging of the containment should be minimized by containment atmosphere clean up systems within the containment." The amount of time that these valves are credited with being open for the purpose of maintaining airborne activity concentrations should be consistent with the Probabilistic Risk Assessment used for containment openings. Because information provided in the applicant's response was inconsistent with other information in the design basis documents, **RAI 427-2909, Question 12.02-19**, was considered closed but the issues it raised remained open and the staff issued follow-up **RAI 532-4019, Question 12.02-25**, asking the applicant to provide justification for allowing the containment purge isolation valve to remain open on a continuous basis, or to revise the responses to **RAI 143-1737, Question 12.02-10**, dated February 6, 2009, and **RAI 427-2909, Question 12.02-19**, dated September 28, 2009. The applicant's response to **RAI 532-4019, Question 12.02-25,** dated April 9, 2010, committed to revising Table 12.2-60 "Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Containment) (Sheet I of 3)" to indicate that the purge valve will only be intermittently opened in support of CB entries. Because the applicant's response conforms to the guidance contained within BTP 6-4 regarding containment integrity and the SRP guidance regarding defining airborne radioactive sources, **RAI 532-4019, Question 12.02-25,** was resolved. However, DCD Tier 2, Revision 2, Subsection 12.2 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 532-4019, Question 12.02-25,** is identified as **Confirmatory Item 12.02-3** and the staff will confirm that this information is included in a future revision of the DCD.

While the confirmatory calculations for airborne activity concentrations performed by the staff are in close agreement with the data provided by the applicant, the NRC staff noticed that the fractional Derived Airborne Concentrations (DAC) airborne concentration exceeded one DAC in ventilation zones V-VI (RB and A B) and the CB. These estimated airborne activity concentrations are not consistent with the guidance of SRP Section 12.3-12.4 Acceptance Criteria 3, which notes that the ventilation system is to have adequate capability to reduce concentrations of airborne radioactivity to one DAC, in areas not normally occupied where maintenance or in service inspection must be performed. The staff issued **RAI 532-4019, Question 12.02-26**, asking the applicant to reconcile the calculated airborne activity concentrations with the guidance provided in the SRP. The applicant's response to **RAI 532-4019, Question 12.02-26**, dated April 9, 2010, provided a proposed revision to Subsection 12.2.2 that states that if entry is needed to an area that has elevated airborne activity concentrations, appropriate personnel stay times and protective equipment would be utilized to maintain personnel doses in compliance with 10 CFR Part 20. The applicant's proposed alternative provides an acceptable method of complying with the relevant NRC requirements contained in 10 CFR 20 Subpart H for controlling internal exposure from airborne radioactivity which states that when it is not practical to apply process or other engineering controls to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, the licensee can control access, limit access time, use respiratory protection equipment or employ other controls to maintain dose ALARA. Since the applicant's response meets the requirements specified in 10 CFR 20 Subpart H, **RAI 532-4019, Question 12.02-26,** was resolved. However, DCD Tier 2, Revision 2, Subsection 12.2 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 532-4019, Question 12.02-26,** is identified as **Confirmatory Item 12.02-4** and the staff will confirm that this information is included in a future revision of the DCD.

As discussed above, the applicant's response to **RAI 427-2909, Question 12.02-19**, dated September 28, 2009, provided ventilation system parameters affecting airborne activity concentrations, and also included the assumptions regarding fluid stream flow rates which also affect the source term available to become airborne. However, in its response to **RAI 427-2909, Question 12.02-19**, dated September 28, 2009, the applicant provided system flow rate data that was not consistent with the component design information provided in DCD Tier 2, Revision 2, Section 9.3.4 "Chemical and Volume Control System". The portion of **RAI 427-2909, Question 12.02-19,** related to CVCS system flow rate data was therefore closed but the issue it raised remained open. Therefore, the staff issued follow-up **RAI 532-4019, Question 12.02-27**, requesting that the applicant revise DCD Tier 2, Section 12.2 of the DCD to use flow rate data consistent with the plant design basis. The applicant's response to **RAI 532-4019, Question 12.02-27,** dated April 9, 2010, acknowledged the inconsistent information and provided a revised CVCS purification flow rate. However, this revised CVCS system flow rate also exceeded the flow rate capacity of the relevant CVCS system components. Following a teleconference with the applicant on August 05, 2010, to discuss this inconsistent information, the applicant submitted a revised response to **RAI 532-4019, Question 12.02-27, dated September 14, 2010**, proposing a change to DCD Subsection 9.3.4.1.2.3 which states that the CVCS system is capable of providing purification flow rates consistent with their assumed flow rate, when using the RHRS for letdown during shutdown. However, because the applicant's response does not address how this assumed flow is purified, this issue is not resolved. **RAI 532-4019, Question 12.02-27,** is associated with the above request and the staff identified this as **Open Item 12.02-1.**

Airborne radioactivity concentrations can occur in containment, both during power operation (coolant leakage) and refueling (evaporation of the refueling pool). The CVCS and the RHRS are designed to provide the capability to purify the reactor coolant through the purification demineralizer following reactor shutdown and cool down. This mode of operation will ensure that the effect of activity spikes does not significantly contribute to the containment airborne activity during refueling operations. Airborne radioactivity concentrations within the RB and AB result principally from equipment leakage. The design of the plant ensures that expected airborne isotopic concentrations in all normally occupied areas resulting from realistic source terms, are well below the derived air concentration listed in 10 CFR 20 Appendix B to Part 20 "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage". If entry is needed in areas where airborne concentrations exceed the limit (such as containment during normal operation), occupancy time will be adjusted. Under normal operating conditions (no primary-to-secondary leaks and no leakage into the CCWS), components within the uncontrolled portions of the AB are not expected to contain high levels of radioactive material. Airborne radioactivity concentrations in the un-controlled portions of the AB are therefore expected to be negligible. The ventilation from these areas is exhausted through the radiological plant ventilation system, which is monitored by the Plant Vent Stack radiation monitor.

Under normal operating conditions (no primary-to-secondary leaks and no leakage into the CCWS), components within the TB are expected not to contain radioactive material, other than tritium. Airborne radioactivity concentrations in the TB are therefore expected to be negligible under conditions of no leakage. For the purpose of the design basis evaluation, the radioactivity in the main steam system is based on a SG tube leakage rate of 150 gallons per day concurrent with a one percent failed fuel fraction. Continuous operation with primary-to-secondary leakage is assumed. The RCS radionuclide concentrations used are those tabulated in DCD Tier 2, Revision 2, Table 11.1-2. DCD Tier 2 Table 11.1-8 “Parameters Used to Describe Realistic Sources” describes the source parameters that allow calculation of the expected secondary side activity values for a nominal primary to secondary leakage rate of 75 pounds per day (lbm/d). The expected secondary activity values are depicted in DCD Tier 2, Table 11.1-9 “Realistic Source Terms”. Despite the TS allowable leakage rate of up to 600 gallons/day through all SG, and 150 gallons/day through any one SG, actual primary to secondary leakage rates are expected to be maintained low. The use of realistic source terms for describing secondary coolant system corrosion and activation product source terms are based on operating experience from reactors of similar design, which conforms to the guidance contained in the SRP. A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated release from the system, is addressed in Chapter 11. Because the airborne activity concentrations in the TB are expected to be negligible, the TB ventilation system does not contain any radiation monitors. Based on operating experience from reactors of similar design operating with primary to secondary leakage within TS limits, and the guidance contained in RG 8.25, which states that continuous air monitoring should be provided if there is a potential for intakes to exceed 40 DAC hours in 1 day, after taking credit for respiratory protection, this is acceptable because it conforms to the guidance contained in the SRP.

12.2.5 COL Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the DCD:

Table 12-2
US-APWR Combined License Information Items

Item No.	Description	Section
12.2(1)	The COL applicant is responsible for the use of any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography.	12.2
12.2(2)	The COL applicant is to provide the detailed design of additional storage space for radwaste and/or additional radwaste facilities for dry active waste.	12.2

COL information items not identified in Table 1.8-2 of the DCD: None

The identified COL information items address information that is site-specific and beyond the scope of the staff’s review of the DCD. The staff has reviewed COL Information Items 12.2(1) and 12.2(2), and finds that they are relevant, complete, and appropriate for this section and focused on matters that may be a significant issue in any COL application referencing the DCD, and therefore, conform to the guidance contained in the SRP.

12.2.6 Conclusions

The applicant has provided a description of contained and airborne radioactivity sources used as inputs for the dose assessment and for shielding and ventilation designs. The applicant also included the assumptions used in arriving at quantitative values for these contained and airborne source terms, based on the guidance of SRP Section 12.2 (NUREG-0800), or justified appropriate alternative methodologies. For post-accident shielding for vital area access, the applicant used the source terms in NUREG-0737 and RG 1.183.

During power operation, the greatest potential for personnel dose is inside the containment from N-16, noble gases, and neutrons. Outside the containment, and after shutdown inside the containment, the primary sources of personnel exposure are fission products from fuel clad defects and activation products, including activated corrosion products. The coolant and corrosion activation products are based on operating experience data. Neutron and prompt gamma source terms are based on reactor core physics calculations and operating experience from reactors of similar design. DCD Tier 2, Section 11.1, “Source Terms,” contains other parameters used, as well as a complete description of the routine operational source term development.

The main sources of airborne radioactivity are from sources located inside containment and the RB. Leakage from the equipment constitutes the main airborne sources for the CB. Leakage from equipment is the main source of airborne concentrations in the RB and AB. The applicant has provided a tabulation of the maximum expected routine radioactive airborne concentrations for normally occupied areas. Operational experience has shown airborne radioactivity to be a negligible contribution to personnel dose when using realistic source terms, and the US-APWR design ensures that any airborne radioactivity is contained and reduced through the use and design of the CB, RB and AB ventilation systems. Features such as the use of pressure gradients to direct air flow ensures that airborne radioactivity levels will be negligible in the normally occupied areas of the plant.

The staff has reviewed the applicant’s submittal to the requirements of 10 CFR Part 20 as it relates to limits on doses to occupationally exposed persons in restricted areas, and the requirements of GDC 61, as it relates to the information on radiation sources provided by the applicant.

Except for the matters identified in Confirmatory Items:

RAI 427-2909 Question 12.02-21	**Confirmatory Item 12.02-1**
RAI 532-4019 Question 12.02-23	**Confirmatory Item 12.02-2**
RAI 532-4019 Question 12.02-25	**Confirmatory Item 12.02-3**
RAI 532-4019 Question 12.02-26	**Confirmatory Item 12.02-4**
RAI 532-4019 Question 12.02-28	**Confirmatory Item 12.02-5**
RAI 532-4019 Question 12.02-30	**Confirmatory Item 12.02-6**
RAI 561-4441 Question 12.02-31	**Confirmatory Item 12.02-7**

and **Open Items**:

- **RAI 532-4019, Question 12.02-27** **Open Item 12.02-1**
- **RAI 532-4019, Question 12.02-29** **Open Item 12.02-2**

The staff finds, for the reasons set forth above, that the applicant's description of contained and airborne sources complies with the requirements of 10 CFR Part 20 and GDC 61.

SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Application Without Inspections, Tests, Analyses, and Acceptance Criteria" states that in the absence of ITAAC, "fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability at the COL stage. The DCD specifies that the COL applicant will have a radiation protection program that meets the requirements of NEI 07-03A,and an ALARA program that meets the requirements of NEI 07-08A "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)", which provide acceptable methods of describing the radiation protection and ALARA programs. The staff has reviewed NEI 07 03A (ML091490684) and NEI 07-08A (ML093220178) and determined them to be acceptable. Since the radiation protection program is used to assess and control the receipt, storage and use of radioactive sources, COL Information Item 12.2(1) conforms to the guidance contained in the SRP for the identification of miscellaneous sources, and is therefore acceptable.

12.3 and 12.4 Radiation Protection Design Features (including Dose Assessment)

This section is written to cover Tier 2, Section 12.3 and 12.4 because NUREG-0800 Section 12.3-12.4 is written to cover both sections.

12.3.1 Introduction

This section focuses on radiation protection design features, including the equipment used for assuring that ORE will be as low ALARA. Dose rates during normal operation, AOOs, and accident conditions are considered. Radiation zones are defined for various modes of plant operation. Design features to control personnel radiation exposures include the physical layout of equipment, shielding and barriers to high radiation areas, fixed area radiation, and continuous airborne radioactivity monitoring instrumentation, including instrumentation for accident conditions. The estimated annual personnel doses associated with major functions, such as operation, handling of radioactive waste, normal maintenance, special maintenance (e.g., SG tube plugging), refueling, and in-service inspection provide a measure of the effectiveness of the proposed design features, in reducing overall area dose rates.

12.3.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1, Sections 2.7.6.2, 2.7.6.4, 2.7.6.6, 2.7.6.7, 2.7.6.13 and 2.8, and consists of design features which demonstrate compliance with the occupational radiation safety requirements of 10 CFR Part 20 including those Tier 1 sections which address; dose significant shield walls located in the Nuclear Island and Radioactive Waste Buildings, the containment high radiation accident monitors, the MCR ventilation accident radiation monitors, and the nuclear island ventilation system. In addition these sections identify relevant Key Design Features, seismic classifications, interlocks, Class 1E power supplies, and ITAAC.

DCD Tier 2: The applicant has provided a DCD Tier 2, system description in Section 12.3-12.4, summarized here in part, as follows:

Radiation protection design features include shielding, ventilation, radioactivity monitoring systems, and contamination control. Also presented in this section of the US-APWR application is a projected annual personnel dose assessment for the US-APWR.

The inner compartment of the CB contains the SGs, reactor coolant pumps, and primary loop piping. The CB outer compartment houses support equipment. Shielded compartments are provided for CVCS components located outside of the secondary shield in containment. Plant personnel do not routinely enter CB during power operations.

A hot machine shop is provided for receiving, disassembling, repairing, and machining activated or contaminated components to control the spread of contamination and provide a low dose rate area for servicing. A hot tool storage area adjacent to the hot workshop is provided for the control, storage, issuance, and receipt of contaminated tools and equipment. Radioactive piping and associated equipment are isolated and drained for routine maintenance.

Ventilation provisions to protect workers from airborne radioactive material include air pressure gradients from low potential airborne contamination areas to areas of higher potential airborne contamination and then exhaust of the air through filters.

Very high radiation areas in the CB during normal and refueling operations include the reactor cavity, the core internals storage area, the instrument lance storage, and the fuel transfer pit. The very high radiation areas in the fuel handling areas of the RB during normal and refueling operations include the Fuel Transfer Tube inspection area the Fuel Transfer Pit, the SFP, Fuel Inspection Pit and the Cask Loading Pit. These areas are expected to be flooded with water or otherwise inaccessible by workers when very high radiation levels are present.

Radiation zones for each plant area are defined by the dose rate in the areas taking into account sources within each area as well as contributing dose rate from sources in adjacent areas. Radiation zone categories are described in Table 12.3-2 - US-APWR Radiation Zone Designation.

The area radiation monitoring instrumentation for use during normal operation and abnormal operating occurrences is provided to measure the radiation levels in specific areas of the plant and to create a continuous record of radiation levels at key locations. The instrumentation also warns of possible equipment malfunctions, and leaks in specific areas and furnishes information to supplement radiation surveys. The area radiation monitoring instruments for routine monitoring are powered by a Non-1E power supply, which is backed up by the Non- 1E Alternate AC gas turbine generator. Area radiation monitoring equipment used during postulated accidents is powered by the Class 1E, 120-Vac buses. Monitors designated as safety-related are part of the safety-related portion of the Protection and Safety Monitoring

System and are designed for redundancy, diversity, and independence in accordance with the Institute of Electrical and Electronics Engineers (IEEE) Standards.

Radiation exposures to facility personnel result primarily from direct gamma radiation from components and equipment containing radioactive material. Experience at operating LWR indicates that any dose from airborne radioactivity will not be a significant contribution to the total dose. The applicant estimates a total annual ORE for a US-APWR unit of 0.7103 person-Sv (71.03 person-rem). This includes the activities of reactor operations and surveillance, routine maintenance, in-service inspection, special maintenance (such as SG re-tubing), waste processing, and refueling.

ITAAC: The ITAAC associated with DCD Tier 2 Section 12.3 and 12.4 are given in DCD Tier 1, Sections 2.7.6.2, 2.7.6.4, 2.7.6.6, 2.7.6.7, 2.7.6.13 and 2.8.

TS: TS for the control of high radiation areas are addressed in DCD Tier 2, Chapter 16, TS, Section 5.7, “High Radiation Area.” TS for post accident monitoring (PAM) instrumentation are addressed in DCD Tier 2, Chapter 16, Section 3.3.3, “Post Accident Monitoring (PAM) Instrumentation.”

COL information or action items - (See Subsection 12.3.5 below).

Technical Report(s): There are no technical reports associated with this area of review.

Topical Report(s): There are no topical reports associated with this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues associated with this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements associated with this area of review.

Cross-cutting Requirements (Three Mile Island [TMI], Unresolved Safety Issue [USI]/Generic Safety Issue [GSI], Op Ex): TMI issues 10 CFR 50.34(f)(2)(xxvii) [NUREG 0737 Item III.D.3.3], 10 CFR 50.34(f)(2)(vii) [NUREG 0737 Item II.B.2] and 10 CFR 50.34(f)(2)(viii) [NUREG 0737 Item II.B.2.]. In addition, there is information pertinent to TMI action items in Section 1.9.3, Table 1.9.3-2.

12.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 12.3-12.4 of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 12.3-12.4 of NUREG-0800.

1. 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, as they relate to persons involved in licensed activities making every reasonable effort to maintain radiation exposures ALARA.

2. 10 CFR 20.1201, as it relates to occupational dose limits for adults.

3. 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, 10 CFR 20.1204, 10 CFR 20.1701, “Use of process or other engineering controls,” and 10 CFR 20.1702, “Use of other controls,” as they relate to design features, ventilation, monitoring, and dose assessment for controlling the intake of radioactive materials.

4. 10 CFR 20.1301 and 10 CFR 20.1302, “Compliance with dose limits for individual members of the public,” as they relate to the facility design features that impact the radiation exposure to a member of the public from non-effluent sources associated with normal operations and AOO.

5. 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and the generation of radioactive waste.

6. 10 CFR 20.1601, “Control of access to high radiation ares,” 10 CFR 20.1602, “Control of access to very high radiation areas,” 10 CFR 20.1901, “Caution signs,” 10 CFR 20.1902, “Posting requirements,” 10 CFR 20.1903, “Exceptions to posting requirements,” and 10 CFR 20.1904, “Labeling containers,” as they relate to the identification of potential sources of radiation exposure and the controls of access to and work within areas of the facility with a high potential for radiation exposure.

7. 10 CFR 20.1801, as it relates to securing licensed materials against unauthorized removal from the place of storage.

8. 10 CFR Part 50, Appendix A, GDC 19, “Control room,” as it relates to the provision of adequate radiation protection to permit access to areas necessary for occupancy after an accident, without personnel receiving radiation exposures in excess of 50 millisievert (mSv) (5 rem) to the whole body or the equivalent to any part of the whole body for the duration of the accident in accordance with 10 CFR 50.34(f)(vii).

9. 10 CFR Part 50, Appendix A, GDC 61, as it relates to occupational radiation protection aspects of fuel storage, handling, radioactive waste, and other systems that may contain radioactivity, designed to ensure adequate safety during normal and postulated accident conditions, with suitable shielding and appropriate containment and filtering systems.

10. 10 CFR Part 50, Appendix A, GDC 63, “Monitoring Fuel and Waste Storage,” as it relates to detecting excessive radiation levels in the facility.

11. 10 CFR 50.68, “Criticality accident requirements,” as it relates to procedures and criteria for radiation monitoring in areas where special nuclear material is stored and handled.

12. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the COL, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

The following RGs, NUREGs, and industry standards provide information, recommendations, and guidance and in general describe a basis acceptable to the staff for implementing the requirements of the regulations identified above:

1. RG 1.7, Revision 3, "Control of Combustible Gas Concentrations in Containment," as it relates to methods for determining gaseous radionuclides in containment following an accident.

2. RG 1.52, Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants", as it relates to radiation protection considerations for engineered safety feature atmosphere cleanup systems operable under postulated DBA conditions, to be designated as "primary systems."

3. RG 1.69, Revision 0, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants", as it relates to the requirements and recommended practices acceptable for construction of facilities that apply to occupational radiation protection shielding structures for nuclear power plants.

4. RG 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants", as it relates to a method acceptable to the staff for complying with the Commission's regulations to provide instrumentation for radiation monitoring following an accident in a light-water-cooled nuclear power plant.

5. RG 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," as it relates to the assumptions and methods for evaluating doses to individuals accessing the facility during and following an accident in accordance with NUREG-0737, Item II.B.2.

6. RG 8.2, Revision 2, "Guide for Administrative Practices in Radiation Monitoring", as it relates to general information on radiation monitoring programs for administrative personnel.

7. RG 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable", as it relates to actions taken during facility design, engineering, construction, operation, and decommissioning to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, concerning the radiation protection information to be supplied in Safety Analysis Report, Section 12.

8. RG 8.10, Revision 1-R "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable", as it relates to the commitment by management and vigilance by the radiation protection manager and staff to maintain ORE ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003.

9. RG 8.38, Revision 1, "Control of Access to High and Very High Radiation Areas of Nuclear Plants", as it relates to the physical controls for personnel access to high and very high radiation areas.

10. ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors," as it relates to criteria for the establishment of locations for fixed continuous area gamma radiation monitors and for design features and ranges of measurement.

11. ANSI N13.1-1999, "Sampling and Monitoring Releases of Airborne Radioactive Substances From the Stacks and Ducts of Nuclear Facilities," as it relates to the principles that apply in obtaining valid samples of airborne radioactive materials, and acceptable methods and materials for gas and particle sampling.

12. ANSI/ANS-6.4-1997 (R2004), "Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants", as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures.

12.3.4 Technical Evaluation

The staff reviewed the radiation protection design features, dose assessment, and minimization of contamination design considerations contained in DCD Tier 2, Section 12.3, for consistency with the guidance in RG 1.206 and the criteria in Section 12.03-12.04 of the SRP. The purpose of this review was to ensure that the applicant had either committed to follow the guidance of the RGs and applicable staff positions, or offered acceptable alternatives. Where the DCD is consistent with the guidance in these RGs and staff positions, the staff can conclude that the relevant requirements of 10 CFR Parts 20, 50, and 70 have been met. The following sections present the staff's findings.

12.3.4.1 Radiation Protection Design Features

The reactor design incorporates features to help maintain occupational radiation exposures ALARA in accordance with the guidance in RG 8.8. These include facility design, shielding, ventilation, and area and airborne radiation monitors. These design features are founded on the ALARA design considerations described in DCD Tier 2, Section 12.1 and discussed in Section 12.1 of this SE.

12.3.4.1.1 Facility Design Features

The main sources of radiation are the reactor vessel, the primary loop components and associated piping. The primary shield, in conjunction with the secondary shield, reduces radiation levels from the reactor components and primary loop components and piping, to levels that allow limited access during operation. Exposure rates and component activation are reduced by the incorporation of labyrinth shielding in the design of the structure around the reactor vessel.

The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. Insulation, in the area of the reactor vessel nozzle welds, is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate the removal of the insulation for the inspection of the welds.

The Reactor Coolant Pump (RCP) design includes the use of an assembled cartridge seal for the Number 2 and Number 3 pump seal that reduces the time required for replacement. The RCP design also includes a spool piece to facilitate the assembly or disassembly of the seal system without the removal of the motor from the pump.

The SG design incorporates features to facilitate maintenance and inspection that reduce ORE, including: sizing the primary side channel head manway openings to facilitate access, provision of a cylindrical region just below the tube sheet to facilitate access to the outer rows of SG tubes, and a nozzle dam design that reduces the time for installation and removal. As described in DCD Tier 1, Revision 2, Section 2.4.2 "Reactor Coolant System" the SG tube material is thermally treated alloy 690. The specification of Alloy 690 tubing material improves equipment reliability, and the low cobalt content results in lower radiation exposure rates in the SG channel heads, and throughout the plant.

While DCD Section 12.1.2.1 notes that the use of low cobalt material and provision of features to prevent buildup of radioactive material are effective methods of reducing personnel exposure, specifications of allowable cobalt impurities in primary plant construction materials, was not provided in DCD Tier 2, Revision 1, Section 12.3. This information was also not provided in DCD Tier 2, Revision 1, Section 5.2.3 "Reactor Coolant Pressure Boundary Materials", DCD Tier 2, Revision 1 "Auxiliary Systems" or DCD Tier 2 Chapter 6 "Engineered Safety Features". Reactor operating experience provided in standard industry documents, notes that limiting cobalt impurity levels in materials used to construct the RCS, reactor vessel and internals could significantly reduce long term plant dose rates. Guidance contained in some of these documents recommends that the cobalt impurity in stainless steels should be less than 500 ppm, and Inconel alloys should have a cobalt content of less than 200 ppm. The cobalt content of nickel plate used to fabricate in-core components should be less than 50 ppm and braze material should be less than 500 ppm. Because the DCD did not address these known, proven and documented dose reduction techniques, in **RAI 147-1850, Question 12.03-12.04-4**, the staff asked the applicant to include the information that describes the design specifications for the material selection employed for the purpose of dose reduction ALARA. The applicant's response to **RAI 147-1850, Question 12.03-12.04-4**, dated February 6, 2009, provided specifications for components exposed to high temperature reactor coolant including limitations on the cobalt content of the base metal as given in DCD Tier 2 Table 12.3-7 "Equipment Specification Limits for Cobalt Impurity Levels". The use of hard facing material with Co content such as stellite is limited to applications where its use is necessary for reliability considerations. The applicant also noted that Nickel-based alloys in the RCS (Co-58 is produced from activation of Nickel (Ni)-58) are similarly used only where component reliability may be compromised by the use of other materials. Because radiation from Co-60 accounts for greater than 90 percent of ORE at commercial nuclear power plants, the cobalt content of materials exposed to high neutron flux should be limited. Industry experience shows that residual cobalt contamination is the main source of radiation from activated structural components. Due to their proximity to high neutron flux, cobalt impurities from the core barrel and shroud were major sources of activation of material that resulted in increased ORE, and increases in decommissioning expenses. In 10 CFR Part 20 the definition for ALARA includes guidance to make every reasonable effort to maintain exposures below regulatory limits, taking into account the state of technology. In addition, the provisions in 10 CFR 20.1406(b) requires applicants to describe how the facility design will minimize, to the extent practicable, contamination of the facility and, facilitate eventual decommissioning. Therefore, **RAI 147-1850, Question 12.03-12.04-4,** is considered closed, but the issues remain open, and in follow up question **RAI 428-2910,**

Question 12.03-12.04-24, the staff asked the applicant to explain how the allowed use of higher cobalt materials located in high neutron flux areas was consistent with ensuring ORE is ALARA and decommissioning costs are minimized in accordance with the requirements in 10 CFR 20. While the applicant's response to **RAI 428-2910, Question 12.03-12.04-24**, dated September 28, 2009, stated that the applicant's specifications for allowable cobalt content were consistent with industry guidance, the staff noticed that contrary to the information provided by the applicant, the current industry recommendations were less than the values stated by the applicant. Since the specification for allowable cobalt impurities provided by the applicant in DCD Tier 2 Table 12.3-7 will potentially result in increased ORE and higher facility contamination the staff concluded that **RAI 428-2910, Question 12.03-12.04-24,** is considered closed, but the issues remain open, and therefore issued follow-up **RAI 524-4020, Question 12.03-12.04-34**, which requested that the applicant justify the use of cobalt impurity limitations that exceed the recommendations provided in industry literature. The applicant's response to **RAI 524-4020, Question 12.03-12.04-34**, dated March 12, 2010, stated that while the specifications adopted by the applicant exceeded the industry recommended cobalt content values for some components, due to lower allowable cobalt content in other components, the overall cobalt introduction rate was less than that recommended in industry literature, and was therefore acceptable. The NRC staff identified that the calculations for the cobalt introduction rate provided by the applicant were erroneous, and that the actual cobalt introduction rate was greater than that recommended by industry. Following a teleconference with the applicant on May 26, 2010, the applicant submitted a revised the response to **RAI 524-4020, Question 12.03-12.04-34,** dated October 8, 2010, which changed the allowable cobalt content of materials used in the RCS to values consistent with industry recommendations. Accordingly, the staff finds the applicant's response conforms to the guidance in the SRP for design features to maintain doses ALARA, and the staff concludes that **RAI 524-4020, Question 12.03-12.04-34,** is resolved. However, DCD Tier 2, Revision 2, Subsection 12.1.1.1.1 and Table 12.3-7 have not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 524-4020, Question 12.03-12.04-34,** is identified as **Confirmatory Item 12.03-12.04-8** and the staff will confirm that this information is included in a future revision of the DCD.

US-APWR DCD Tier 2, Section 12.3.1 states that filters and demineralizers are provided as part of the plant design. Liquid systems containing radioactive cartridge filters are provided with a remote filter handling system for the removal of spent radioactive filter cartridges from their housings for packaging and shipment for burial. The process is accomplished so that the exposure to personnel and the possibility of an inadvertent radioactive release to the environment are minimized. Each filter is provided with a vent and a drain valve and is contained in a shielded compartment provided with drainage capability. The filter handling system has also been designed with a minimum of components susceptible to malfunction. Industry standard documents note that the benefits of using submicron filters may include lower coolant activity that leads to easier decontamination and reduced contamination levels, lower general area dose rates fewer hot spots and improved RCP seal life. For an operating plant, the result of improved seal life would be less work and dose associated with seal maintenance and replacement and improved plant reliability. This method of material specification is a known, proven, cost effective and documented dose reduction technique. DCD Tier 2, Section 12.3.1.1.1.2 and DCD Tier 2, Section 9.3.4.2.6 discuss nuclear plant filters, but neither section provides any performance specification requirements for these filters. In **RAI 170-1856, Question 12.03-12.04-6**, the staff asked the applicant to provide performance specification requirements for these filters. The applicant's response to **RAI 170-1856, Question 12.03-12.04-6**, dated March 4, 2009, acknowledged the benefit of improved RCP seal life, but stated that the potential for an increased number of filter replacements, and the associated increases in ORE and waste, were factors that should be addressed during plant operation. The adoption of particular dose reduction strategies is dependent on plant specific operational considerations that are addressed in the ALARA program. The plant staff selection of filter media specifications is an operational issue that conforms to the guidance contained in RG 8.8 for maintaining occupational radiation exposures ALARA in accordance with 10 CFR 20.1101(b). Accordingly, the staff finds the applicant's response acceptable and concludes that **RAI 170-1856, Question 12.03-12.04-6,** is resolved and closed.

Demineralizers for highly radioactive systems are designed to minimize personnel doses. The demineralizer spent resins are remotely and hydraulically transferred to the spent resin storage tanks (SRST) so that fresh resin can be remotely loaded into the demineralizer. The demineralizers and piping are designed with the ability to be flushed with demineralized water. Strainers are installed in the vent lines to prevent the entry of spent resin into the exhaust duct.

The US-APWR design uses BAE to process RCS and reuse the concentrated boric acid (BA) and the BAE distillate output in the CVCS reactivity control system. The concentrated BA output of the BAE is pumped to the Boric Acid Storage Tanks (BAST) and the Boric Acid Transfer Pumps (BATP) located in the AB. The DCD states that adequate space and flanged connections for easy removal are provided for the maintenance of evaporator components. The evaporator can be run in an automatic operation mode to reduce the exposure of the operator to radiation from the equipment. DCD Tier 2, Revision 1, Figure 12.3-1 "Radiation Zones for Normal Operation/Shutdown (Sheet 17 of 34) Auxiliary Building at Elevation 3'-7"," identifies the BAE as a Zone VIII (Maximum Dose Rate 100 Roengten/hour) area. Based on the information depicted in this figure, two pumps are located next to the evaporator. DCD Section 12.3.1.1.2 "Common Facility and Layout Designs for As Low As Reasonably Achievable," Subsection E. "Equipment Layout," notes that for major radiation sources, such as the Boric Acid Recycle System, equipment separation is used as a dose reduction technique. However, based on staff experience and documented industry experience work on components located adjacent to the BAE package may result in high personnel external exposure and internal depositions for maintenance personnel working on the interior of the evaporator package. In **RAI 171-1858, Question 12.03-12.04-7,** the staff asked the applicant to describe the dose reduction features of the BATP and BAST which are used to transfer and store the concentrated discharge BA of the BAE, and in **RAI 171-1858, Question 12.03-12.04-8,** and the staff asked the applicant to describe the dose reduction features of the BAE package. The applicant's response to **RAI 171-1858, Question 12.03-12.04-8**, dated March 3, 2009, referred to the response to **RAI 171-1858, Question 12.03-12.04-7**, dated March 3, 2009, which stated that the RCS passes through the MB demineralizer inlet filter, the reactor coolant filter and the BAE feed demineralizer filter before it flows into the boron recycle system. As these components remove suspended matter, no radioactive deposits will remain inside the boron recycle system, and as a result, the dose rates will be insignificant for necessary maintenance activities. Therefore, no partition is included between the two pumps as it would reduce the amount of work space. However, based on the operating experience of the NRC staff personnel, corroborated by industry literature, some plants, operating with significantly fewer fuel defects than assumed in the US-APWR DCD, have experienced high dose rates on BAE related components. The applicant's assumption that no activity would be present in the process fluid streams after the BAE feed stream demineralizers and filters is inconsistent with the NRC staff operating experience and is inconsistent with the applicant's response to **RAI 168-1739, Question 12.02-14,** dated March 4, 2009, evaluated in Section 12.2, which provided DF and concentration factors that are consistent with the experience of the staff. Operation of the BAE package with the stated removal and concentration factors would not only result in appreciable amounts of radioactive material being present downstream of the BAE package under normal conditions, but especially when operating with cladding defects. The use of a source term based on operational experience appears to be contrary to the source term stated in DCD Section 12.2.1.1 "Sources for Full-Power Operation," which states that the design basis for the shielding is the source term resulting from full-power operation with cladding defects in the fuel rods producing one percent of the core thermal power. The use of cladding defects for determining the shielding requirements is consistent with the acceptance criteria contained in SRP Section 12.2. Therefore, **RAI 171-1858, Question 12.03-12.04-7,** and **RAI 171-1858, Question 12.03-12.04-8,** are considered closed, but the issues remain open. In follow up **RAI 428-2910, Question 12.03-12.04-23**, the staff asked the applicant to describe the removal factors, and their bases, assumed for the responses to questions **RAI 171-1858, Question 12.03-12.04-7**, dated March 3, 2009 and **RAI 171-1858, Question 12.03-12.04-8**, dated March 3, 2009. Provide justification for the use removal factors other than those noted in the response to **RAI 168-1739, Question 12.02-14,** dated March 4, 2009, and describe the design feature provided to reduce ORE associated with the BAE package and downstream components, and the source term used as the basis for providing those design features. The applicant's response to **RAI 428-2910, Question 12.03-12.04-23,** dated September 28, 2009, stated that, while the BAE vault shielding design was based on the assumed design basis activity values, based on their operating experience with realistic source terms, and the use of filters and demineralizers in the feed stream to the BAE package, they did not expect high radiation levels in the vicinity of the BAE package or the Boric Acid Transfer Pumps. The applicant also stated that flanged connections were provided to facilitate the removal of the pumps to lower radiation areas for maintenance and that installation of permanent shielding between the components would reduce the space available to work on the components, which would potentially increase worker exposure. Because the use of realistic source terms for estimating ORE conforms to the guidance in the SRP, the staff concluded that **RAI 428-2910, Question 12.03-12.04-23,** is resolved and closed. **RAI 427-2909, Question 12.02-18,** (a supplemental question derived from **RAI 142-1733, Question 12.02-8** and **RAI 142-1733, Question 12.02-9**), which were evaluated in Section 12.2, provided additional information about activity source terms that would be used to establish ORE associated with maintenance of the BAE units. From the information provided in Table 12.4-1 "Occupational Dose Estimates During Routine Operations and Surveillance" and the source term information provided by the applicant, the NRC staff was unable to ascertain the estimated exposure associated with maintenance and non routine operation and surveillance of the BAE package. Therefore, the staff issued **RAI 532-4019, Question 12.02-24,** which asked the applicant to revise Section 12.2 and Section 12.4 of the DCD to describe the estimated exposure and the associated basis, for routine maintenance and surveillance of the BAE packages. The applicant's response to **RAI 532-4019, Question 12.02-24**, dated April 9, 2010, which was evaluated in Section 12.2, described the applicant's operating equipment experience and current radiation protection program practices that formed the basis for the applicant's assertion that personnel exposure would be ALARA and was adequately described by information already contained in the DCD. The use of relevant experience from operating reactors to establish the basis for source terms conforms to the guidance contained in the SRP, so based on the applicant's response, **RAI 532-4019, Question 12.02-24**, which was evaluated in Section 12.2 was resolved. Because increasing the concentration of fluid in the BAE would also increase general area dose rates around the BAE package and could change the radiation protection measures needed to keep work around the BAE package ALARA, the staff issued **RAI 532-4019, Question 12.02-30**, asking the applicant to describe how the BAE concentration factor would vary and what impact this would have on area dose rates, and related maintenance ORE. The applicant's response to **RAI 532-4019, Question 12.02-30**, dated April 9, 2010, which was evaluated in Section 12.2, stated that the possibility of an increase in radioactive concentration can be seen only during the end of a cycle when the boron concentration is low and the concentration rate in the BAE is high. The applicant committed to adding COL Information Item COL 12.3(9) to confirm the radiation levels in the BAE rooms during the end of cycle instead of changing the source strength by increasing the concentration rate of the BAE. Based on the applicant's response, which was evaluated in Section 12.2, **RAI 532-4019, Question 12.02-30** was resolved. However, DCD Tier 2, Revision 2, Subsection 12.3 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 532-4019, Question 12.02-30** is identified as **Confirmatory Item 12.02-6** and the staff will confirm that this information is included in a future revision of the DCD.

As described in RG 8.8, the sources of radiation that occur in tanks used to process liquids containing radioactive material, such as sedimentation and residual liquids, can be reduced by sloping the tank bottoms toward outlets, so whenever practicable, US-APWR tanks are provided with sloped bottoms and bottom outlet connections. Tank overflow lines are directed to the waste collection system to control any contamination within plant structures. Tanks containing radioactive fluids are either equipped with open vents to the cubicle or the ventilation system.

Some heat exchangers that are large sources of radiation are located in shielded compartments. The US-APWR uses two different types of heat exchangers in systems containing radioactive fluids. Most of the heat exchangers used are the conventional shell and tube type. The tubes are made of corrosion resistant materials to limit corrosion related degradation, and flow impact baffles along with the configuration of the heat exchanger are used to limit flow velocities, thereby reducing impingement wear. The SFP heat exchangers and the component cooling water heat exchangers are plate-type heat exchangers constructed of austenitic stainless steel.

In accordance with the guidance provided in RG 8.8, the US-APWR DCD notes a number of design features that facilitate maintenance and reduce ORE. Plant monitoring and control instruments are located in low radiation zones and away from radiation sources. Some instruments in high radiation zones are duplicated to reduce required access and service time. In the containment, instruments are located outside the secondary shield (the area of lowest radiation at power and during shutdown). Instruments which are located in high radiation zones are designed for easy removal to a lower radiation zone for calibration. Transmitters and readout devices are located in low radiation zones, such as corridors and the control room. Installed check sources are provided for airborne radiation monitors and safety related area radiation monitors response verification. Instrument sensing lines of process piping which may contain highly radioactive solids are equipped with chemical seals to keep the instrument lines free of solids and to reduce servicing time. Instruments and sensing line connections are located slightly above the pipe mid-plane to minimize radioactive activated corrosion products or gas buildup, wherever practical. The staff confirmed that the equipment and layout design features described above conform to the guidance contained in RG 8.8 for maintaining occupational radiation exposures ALARA. Accordingly, the staff finds these features acceptable.

To minimize personnel exposures from the operation of valves, motor-operated, air-operated, or other remotely actuated valves are used, where justified by the activity levels and frequency of use. In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding is provided such that personnel need not enter a high radiation area for valve operation. For equipment which is infrequently operated and located in high radiation areas, all of the manual valves associated with the safe operation of the equipment are provided with remote-manual operators or reach rods. For valves located in radiation areas, provisions are made to drain the adjacent radioactive components when maintenance is required. Some of the valve design features described in US-APWR Tier 2, DCD Sections 12.3.1, Section 5 and Section 9.3, that were intended to improve component reliability or reduce radiation exposure are inconsistent with current operating experience or are incomplete. For example, DCD Tier 2, Section 12.3.1 does not discuss the design specifications applicable to obtaining reliable check valves while minimizing the number of required system intrusive inspections needed to assure reliable operation. The US-APWR DCD only provides limited information regarding pump design specifications provided to reduce maintenance, improve reliability or to reduce leakage. The DCD notes that the RCP design includes the use of an assembled cartridge seal for the Number 2 and Number 3 pump seal that reduces the time required for replacement. The DCD notes that some pumps are sealed with mechanical seals. The design of small pumps includes features to allow easy removal for maintenance in low dose rate areas. Pump casings are provided with drain connections for draining pumps for maintenance. However, a number of industry standard documents, notably those provided by Electric Power Research Institute (EPRI), describe design practices that represent improvements that increase pump efficiency, increase pump reliability, reduce radioactivity build up in the pumps and reduce leakage from the pumps. 10 CFR 20.1101(b) and 10 CFR 20.1406(b) require licensees to describe design features, based on the current state of the technology, that are provided to maintain ORE ALARA, reduce contamination of the facility, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. Also, 10 CFR 52.47 "Contents of applications; technical information" requires applications to include information describing how operating experience has been incorporated into the design. SRP Section 12.3-4 "Acceptance Criteria" and RG 8.8 and RG 4.21, Revision 0, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," provide guidance for meeting the requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406. RG 8.8 notes that doses from servicing valves can be reduced by specifying and installing reliable valves for the required service. A number of industry standard documents, notably those provided by the EPRI, describe changes to previous design practices that have been shown to be erroneous, or which represent improvements over previous design practices. Therefore, the staff issued **RAI 524-4020, Question 12.03-12.04-36**, asking the applicant to update US-APWR DCD Tier 2, Section 12.3.1 to provide component specification information consistent with current industry practices, as described above. The applicant's response to **RAI 524-4020, Question 12.03-12.04-36**, dated March 12, 2010, stated that the component design features for reducing ORE were already evaluated in EPRI report Technical Report-016780, "Advanced Light Water Reactor Utility Requirements Document" (ALWR URD) has been used as the guidance for US-APWR detailed design in addition with other industry standards. The applicant noted that the NRC had issued NUREG-1242 "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document" that evaluated the Utility Requirements Document (URD). However, as noted in NUREG-1242, the URD has no legal or regulatory status. It is not intended to demonstrate complete compliance with the Commission's regulations, regulatory guidance, or policies, nor is it intended to be used as a basis for supporting final design approval (FDA) and DC for a specific design and it specifically noted that the staff's review of the URD is not intended to substitute for any portion of the staff's review of future applications for FDA and DC. The URD was prepared circa 1990 and references a number of industry documents, such as NP- 6516, “Guide for the Application and Use of Valves in Power Plant Systems”, NP-5479, “Application Guidelines for Check Valves in Nuclear Power Plants” and NP-5697, “Valve Stem Packing Improvements”. Over the past fifteen years, the state of the technology has advanced, and revised versions of these reports have been issued. Based on the design information provided in DCD and the applicant’s response, it is not clear to the staff that the applicant has factored this updated information into their design, as required by regulation. In general, the applicant is relying on dated guidance documents to establish design criteria for plant components. This is not consistent with 10 CFR 20.1101(b). Following a teleconference with the applicant on May 26, 2010, which discussed these issues, the applicant submitted a revised response to **RAI 524-4020, Question 12.03-12.04-34,** dated September 14, 2010, which committed to revising the component specification information provided in DCD Section 12.3 to reflect the use of contemporary industry guidance as part of the selection criteria for pumps, valves and other components. Because the information regarding radiation protection design features conforms to the guidance in the SRP, the staff finds the applicant’s response to **RAI 524-4020, Question 12.03-12.04-36,** dated September 14, 2010, resolved. However, DCD Tier 2, Revision 2, Subsection 12.3 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 524-4020, Question 12.03-12.04-36,** is identified as **Confirmatory Item 12.03-12.04-9**, and the staff will confirm that this information is included in a future revision of the DCD.

The US-APWR piping design minimizes ORE due to radioactive material contained in plant piping by limiting personnel proximity to this piping. The applicant achieves this by individually analyzing piping runs to determine the potential radioactivity level and surface dose rate, and then routing the pipes through controlled access areas properly zoned for that level of activity. Field run piping will be minimized wherever possible. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipeways are provided. Whenever practicable, valves and instruments are not placed in radioactive pipe ways. Equipment compartments are used as pipeways only for those pipes associated with equipment in the compartment. Radioactive and non-radioactive piping are separated to minimize personnel exposure. The amount of activity contained in piping that could expose personnel is minimized by (1) providing drains on piping where low points and dead legs cannot be eliminated, (2) prohibiting the use of non-removable backing rings in the piping joints in piping containing radioactive material, (3) making provisions to isolate and drain radioactive piping and associated equipment should maintenance be required, and (4) eliminating activated corrosion product traps in radioactive waste piping utilized for the transport of spent resins or slurries, by using butt welds to the fullest extent possible, ensuring that horizontal runs carrying spent resin are sloped toward the spent resin tanks and using large radius bends instead of elbows. Where sloped lines or large radius bends are impractical, adequate flush and drain capability is provided to prevent flow blockage and minimize activated corrosion product traps. Whenever possible, branch lines having little or no flow during normal operation are connected above the horizontal mid-plane of the main pipe. Piping which carries resin slurries or evaporator bottoms is run vertically as much as possible. To minimize ORE due to pipe maintenance, the piping in pipe chases is designed for the lifetime of the unit. The staff confirmed that the equipment and layout design features described above conform to the guidance contained in RG 8.8 for maintaining occupational radiation exposures ALARA. Accordingly, the staff finds these features acceptable.

Potential exposure of station personnel to radiation from systems containing radiation sources is reduced by means of a station layout that permits the use of distance and shielding between the sources and work locations. In those systems where process equipment is a major radiation source, pumps, valves, and instruments are separated from the process component to allow servicing and maintenance of these items in reduced radiation zones. On some major plant components provision are made for the removal of these components to lower radiation zones for maintenance. In general, control panels are located in low radiation zones. Major components such as tanks, demineralizers, and filters in radioactive systems are isolated in individual shielded compartments. Labyrinth entranceway shields or shielded doors are provided for compartments to prevent radiation streaming or scattering that could cause adjacent zones to exceed dose rate limits for those areas. Some components with high activity content (such as demineralizers, heat exchangers or tanks in the primary coolant system), are located in shielded compartments that are accessed via elevated ladder/stairs or they are located in completely enclosed shielded compartments with hatch openings or removable concrete block walls. For instance, removable blocks have been installed in a wall of the equipment rooms for the residual heat removal pumps and the charging pumps. To minimize unnecessary personnel exposure due to radiation streaming through penetrations, (1) as many penetrations as practicable are located with an offset between the source and the accessible areas, (2) the penetrations are located as far as possible above the floor, (3) labyrinths are used or (4) the area around the penetration is grouted. The staff confirmed that the equipment and layout design features described above conform to the guidance contained in RG 8.8 for maintaining occupational radiation exposures ALARA. Accordingly, the staff finds these features acceptable.

The US-APWR DCD Tier 2, Revision 1, Section 12.3.1 did not discuss the potential impact of plant lighting on ORE. RG 8.8, C2.i notes that adequate lighting is required for safe and efficient operation of the plant and that for lighting in radiation areas, lighting design should include provisions for access to the light fixtures, such as platforms, installed ladders, quick disconnects or movable fixtures. Based on industry experience, some plants have implemented a number of dose reduction methods for normal and emergency lighting. In order to ascertain how plant lighting incorporates the dose reduction guidance contained in RG 8.8, the staff issued **RAI 174-1873, Question 12.03-12.04-12**. The applicant's response to this question, dated February 27, 2009, added Subsection 12.3.1.1.2.G. "Lighting" to the DCD which requires that (1) adequate illumination levels be provided in radiation areas for performing actions required during normal, shutdown, maintenance, and emergency conditions, as described in DCD Tier 2, Subsection 9.5.3, (2) extended service lamps be utilized in high radiation areas to reduce the exposure of station personnel who service the lamps, and (3) wherever possible, design features that permit servicing of the lamps from lower radiation areas be implemented. In order to reduce the potential for ORE, in high radiation areas an emergency lighting system will be provided to permit prompt egress if the station lighting system fails. The staff confirmed that the equipment and design features described above conform to the guidance contained in RG 8.8 for maintaining occupational radiation exposures ALARA. Accordingly, the staff finds these features acceptable. **RAI 174-1873, Question 12.03-12.04-12**, is considered closed.

Radiation zones for each plant area are defined by the dose rate in the areas, taking into account sources within each area as well as contributing dose rate from sources in adjacent areas. Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding. Each room, corridor, and pipe-way of every plant building is evaluated for; potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; maintenance occupancy requirements; general access requirements; and material exposure limits to determine appropriate zoning. All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the requirements of 10 CFR Part 20. Radiation zone categories are described in DCD Tier 2, Table 12.3-2 “Table 12.3-2 Radiation Zones”. Radiation zones shown in Figure 12.3-1 “Radiation Zones for Normal Operation/Shutdown” and Figure 12.3-2 “General Plant Arrangement with Post Accident Vital Areas”, are based upon conservative design data. Actual in-plant zones and control of personnel access will be based upon surveys conducted by the health physics staff, as described in Section 12.5. For those areas of the plant outside the scope of the DCD, COL Information Item 12.03(4), states in part that the COL applicant is to provide the radiation zones on the site-specific plant arrangement plan. The staff found that DCD Tier 2, Revision 1, Figure 12.3-1 “Radiation Zones for Normal Operation/Shutdown (Sheet 16 of 34) Auxiliary Building at Elevation -8'-7”,” identifies several piping areas as Zone IX (Maximum Dose Rate < 500 Rad/h) areas. Section 12.3 of the DCD does not provide any information regarding equipment or components located in these areas that may require access for operation, maintenance or periodic surveillances, such as Motor Operated Valves, Air Operated Valves, process sensors and limit switches. As several of the depicted piping areas are large, and the dose rates are high, equipment located in these areas that may require operation or maintenance should be identified and evaluated for dose reduction provisions. Subsection 12.3.1.1.2 does not discuss any applicable dose reduction design features described in RG 8.8 Position C2.b that could be provided to reduce maintenance and operation related exposure from the equipment or components located in these areas. Also, unlike other areas depicted on the drawings of this elevation and the elevations immediately above and below, design features provided for access control are not provided (i.e. barrier gates) for these piping areas. Some of the piping areas depicted on this drawing are large enough to accommodate personnel access, or contain equipment needing maintenance.

In order to clarify how the plant design maintained ORE ALARA in accordance with 10 CFR 20.1101(b) and the guidance provided in RG 8.8, the staff issued **RAI 171-1858, Question 12.03-12.04-9**, which asked the applicant to identify any components in these areas that would require periodic access, and **RAI 172-1864, Question 12.03-12.04-10**, which requested the applicant to describe the design features provided to prevent unintended personnel access to these areas. The applicant’s responses to these questions dated March 3, 2009, stated that the plant design was being changed to provide additional openings and stairs, beyond those depicted in DCD Figure 12.3-1 Revision 1. While the applicant stated that there were no plans to install equipment in these areas which could require access for operation, maintenance or periodic surveillance tests, they did note that the access to one of these areas would be an entrance through a valve area. Because the applicant’s response to these questions included inconsistent information, **RAI 171-1858, Question 12.03-12.04-9**, and **RAI 172-1864, Question 12.03-12.04-10,** are considered closed but the issues they raised remained open**.** The staff issued follow-up **RAI 428-2910, Question 12.03-12.04-22**, requesting that the applicant clarify its statements and describe the sources of radiation in these areas and the design features provided to reduce ORE in the DCD. The applicant’s response to **RAI 428-2910, Question 12.03-12.04-22**, dated September 28, 2009, stated that the line which has the highest radiation level in the piping room is the spent resin transfer line, and that would cause the area to be classified as Zone IX (< 5 Gy/h [500 Rad/h]). However, confirmatory calculations performed by the NRC staff using some of the resin isotopic concentrations provided in DCD Tier 2, Revision

2, Section 12.2, indicated that dose rates could exceed 5 Gy/h (500 rad/h) at one meter, the criteria for a Very High Radiation Area (VHRA). As discussed in RG 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants”, VHRAs require much stricter controls and physical barriers, since failure to adequately implement effective radiological controls can result in radiation doses that result in a significant health risk. Therefore, **RAI 428-2910, Question 12.03-12.04-22,** is considered closed**,** but the issue it raised remained open, and the staff issued **RAI 524-4020, Question 12.03-12.04-33**, which requested that the applicant provide information regarding the analytical code used to evaluate the dose rates in areas traversed by resin lines and the assumptions and input parameters used to determine the area dose rates due to pipes containing resin. The applicant’s response to **RAI 524-4020, Question 12.03-12.04-33**, dated March 12, 2010, stated that MHI used the source term from the SRST instead of the MB demineralizer. The reasons for using the SRST resin instead of the MB demineralizers resin for the dose assessment are that the frequency of spent resin transport from the MB demineralizer is low, the duration of transport is limited; and the piping area has a locked entrance to strictly control access during normal operation and prohibit access during the transport of spent resins, which is consistent with DCD Subsection 12.3.1.2.1.1 which states that plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy. However, the guidance in SRP Section 12.3 states that the zones should be based on maximum design dose rate in the area. The SRP notes that the dose rate criterion for each of these zones is derived from expected occupancy and access restrictions, which in turn are then used as the basis for the radiation shielding design. Contrary to this guidance, in their response, the applicant noted that instead of using the MB resin activity values, MHI used the SRST resin activity concentrations to determine zone dose rates. Also, there do not appear to be any COL information items describing the methods and procedures required of the COL applicant to limit resin activity concentrations, to prevent the unanticipated generations of a VHRA, while transferring resin. Following a teleconference with the applicant on August 05, 2010, to discuss this inconsistent information, the applicant submitted a revised response to **RAI 532-4020, Question 12.03-12.04-33,** dated September 14, 2010, which committed to changing the DCD Figure 12.3-1 (Sheet 15 of 34), (Sheet 16 of 34), (Sheet 17 of 34), (Sheet 18 of 34) to reflect the use of the MB demineralizer activity as the basis for establishing radiation zones and the additional of clearly define additional access controls to these areas. Because the information regarding source term information conforms to the guidance in the SRP, the staff finds the response acceptable and, **RAI 532-4020, Question 12.03-12.04-33**, was resolved. However, DCD Tier 2, Revision 2, Subsection 12.3 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 532-4020, Question 12.03-12.04-33,** is identified as **Confirmatory Item 12.03-12.04-7**, and the staff will confirm that this information is included in a future revision of the DCD.

US-APWR DCD Tier 2, Revision 1, Section 11.2.1.6 “Mobile or Temporary Equipment” states that a space is provided inside the AB to accommodate the future installation of mobile or temporary equipment. Process and utility piping and electrical connections are provided to forward liquid waste to future mobile system or temporary equipment. The US-APWR DCD Tier 2, Figure 12.3-1 “Radiation Zones for Normal Operation/Shutdown (Sheet 17 of 34) Auxiliary Building at Elevation 3’-7”,” shows an area reserved for a mobile liquid waste processing system (MLWPS) RG 1.206, Section C.1.12.3.2 states that the applicant is to provide the models, codes parameters and assumptions used to demonstrate compliance with RG 1.69, Revision 0, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants”. However, DCD Tier 2, Section 12.3.1, Revision 1, does not provide information regarding what zone level should be used as the basis for installing removable shielding and the required thickness of removable shielding to be provided. Also, the DCD does not provide any COL information items regarding the actions required by the COL applicant for the use of a MLWPS in the area provided. Therefore the staff issued **RAI 262-1972, Question 12.03-12.04-15,** requesting that the applicant provide shielding design information for the MLWPS. The applicant's response to question **RAI 262-1972, Question 12.03-12.04-15**, dated May 7, 2009, provided a note in DCD Table 12.3-1 to use mobile shielding as needed. However, the response to this question was incomplete in that no action items were provided for the COL applicant. Also, the MLWPS is located on the grade level elevation, but no information regarding provisions for prevent of contamination of the facility or the environment, in accordance with 10 CFR 20.1406 and RG 4.21, Revision 0, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning", has been provided. **RAI 262-1972, Question 12.03-12.04-15,** is closed and resolved, with the exception of the portion of **RAI 262-1972, Question 12.03-12.04-15,** related to the issue of MLWPS shielding which remains open, and the staff issued follow up **RAI 429-3178, Question 12.03-12.04-26,** requesting the applicant to describe the design features of the MLWPS provided to minimize contamination of the facility and environment and to identify the actions required by the COL applicant for use of the MLWPS. The applicant's response to question **RAI 429-3178, Question 12.03-12.04-26,** dated September 28, 2009, stated that because the use of the MLWPS system was optional, the applicant was adding COL Information Items 12.3(6), 12.3(7) and 12.3(8) requiring the applicant to address the concerns described above. The requirements placed upon the COL applicant, should they decide to install a MLWPS, are sufficient to address the radiation protection design aspects contained in the guidance provided by the SRP for Chapter 12. However, these COL information items are not yet reflected in US-APWR DCD Tier 2, Revision 2, Section 12.3.6, "Combined License Information". **RAI 429-3178, Question 12.03-12.04-26**, is identified as **Confirmatory Item 12.03-12.04-3** and the staff will confirm that this information is included in a future revision of the DCD.

As described in DCD Tier 2, Revision 2, Section 12.5, the COL applicant is to provide the operational radiation protection program for ensuring that OREs are ALARA, which includes procedures on access control. DCD Tier 2, Revision 2, Section 12.1, notes that this program is to be developed, implemented and maintained as described in the NEI 07-03A. This program satisfies the access control requirements for the control of high radiation areas stated in DCD Tier 2, Chapter 16, "Technical Specifications", Section 5.7, "High Radiation Area", and discussed in DCD Tier 2, Revision 2, Subsection 12.3.1.2.1.2, "Access Control". The programs and procedures associated with access control will be provided by the COL applicant, and are beyond the scope of this review. Any area having a radiation level that could result in an individual receiving an absorbed dose in excess of 5 Gy (500 rad) in 1 hour at 1 meter from the radiation source or from any surface that the radiation penetrates will be posted "Grave Danger, Very High Radiation Area." Measures taken to control access to very high radiation areas will meet the guidance contained in RG 8.38. Physical barriers, sufficient to thwart undetected circumvention of the barrier, provided to control access to VHRA are depicted in DCD Figure 12.3-1.

As required by 10 CFR 50.34(f)(2)(vii), an applicant must fulfill the following requirements:

- Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain radioactive materials.

- Design, as necessary, for adequate access to important areas and for protection of safety equipment from the radiation environment.

Item II.B.2 of NUREG-0737, dated November 1980, provides additional guidance on how these requirements can be met. Item II.B.2 describes source term information that should be used to calculate post-accident radiation levels. Regulations require that the post-accident plant dose rates should be such that the dose to plant personnel should not exceed 0.05 Sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident (per 10 CFR Part 50 and GDC 19). The dose rate in areas that are continuously occupied should be less than 150 micro Sieverts per hour (15 millirem per hour) over 30 days. Item II.B.2 of NUREG-0737 describes a “vital area” as any area that will, or may, be occupied to permit an operator to aid in the mitigation of, or recovery from, an accident. Item II.B.2 also recommends listing all vital areas in the plant, and providing a summary of the integrated doses to personnel for each of the plant areas that are accessed for the duration of the accident. (These doses should include exposure received while in transit between vital areas.) DCD Tier 2, Section 12.3.1.2.2, “Accident Conditions” notes that a radiation and shielding design review has been performed to identify vital areas and equipment. Areas that may require occupancy to permit an operator to aid in the long term recovery from an accident are considered vital. Table 12.3-3 “Mission Dose for the Vital Areas access route after an Accident” lists all of the vital plant areas that may be accessed post accident, as well as associated integrated mission doses. DCD Tier 2, Figure 12.3-2, “General Plant Arrangement with Post Accident Vital Areas (Sheet 1 of 10)” contains plant radiation zone maps which reflect maximum radiation fields over the course of an accident.

The NRC staff identified some areas, such as the RHRS pump cubicles, Safety Injection Pump cubicles, Waste Disposal Panel and Charging Pump cubicles that might require access during post accident conditions, for which mission doses were not provided. The US-APWR DCD Tier 2, Section 12.3.1.2.2, “Accident Conditions” discusses vital area access for long term accident recovery, and provides projected dose rates and mission doses in Table 12.3-3. However, the analysis presented is silent with respect to the assumed airborne activity concentrations, the use of respiratory protection equipment (if any), assumed protection factors (if used) to limit internal exposure, or the use of movable or temporary shielding material to limit external exposure. Therefore, in **RAI 262-1972, Question 12.03-12.04-16**, **(Parts 1 and 3)** the staff asked the applicant to fully describe the analytical method used to determine the total exposure of personnel accessing these areas following an accident and to show mission paths and describe those design exposure assumptions, such as the use of respiratory protection devices, assumed protection factors for respiratory protection devices, required to perform plant operation and monitoring for the duration of the event. The applicant’s response to **RAI 262-1972, Question 12.03-12.04-16,** dated May 7, 2009, **(Parts 1 and 3)** stated that following an accident, it was not necessary to access any areas other than those already identified in the DCD. Portable instruments, including radiation monitors, will be used to accurately determine the airborne activity concentrations in plant areas where personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii)/(vii) and the criteria in Item III.D.3.3 of NUREG-0737. The Operational Radiation Protection Program developed by the COL applicant, as described in DCD Section 12.5, will provide policies and procedures concerning the operation of the plant which will ensure that the radiation exposures are maintained ALARA and will include the definition and description of the monitoring instrumentation and equipment and of other protective equipment (e.g. portable ventilation systems, temporary shielding, etc.). It will also include procedures on radiological surveillance, dose control, and respiratory protection. However the staff believes that it is not reasonable to assume that no operator action is required to realign, maintain monitor equipment that was running prior to the event, or that would be running after the event would not be required. It is also not clear to the staff if any manual operator actions inside radiologically affected areas of the plant are expected as a result of compensatory Emergency Operating Procedure steps. SRP Section 12.3-12.4 A.1 requires compliance with 10 CFR 20.1202, which requires summing of internal and external exposure. GDC 19 requires that the dose to personnel not exceed 5 rem whole body. NUREG-0696 "Functional Criteria for Emergency Response Facilities" explicitly notes that sources of dose for transit between facilities include airborne radioactivity. RG 8.15, Revision 1, "Acceptable Programs for Respiratory Protection", notes that exposure estimates should consider the impact of respiratory protection devices on worker efficiency. Contrary to the requirements of 10 CFR 20.1202, the applicant has not considered exposure from airborne radioactive material as part of the exposure estimate. Therefore, the staff issued **RAI 429-3178, Question 12.03-12.04-27**, **(Parts 1 and 3)** asking the applicant to address these issues. The applicant's response to **RAI 429-3178, Question 12.03-12.04-27,** dated September 28, 2009, stated that in DCD Section 12.3.1.2.2, that the US-APWR vital areas which require personnel access within 30 days after the occurrence of an accident are limited to the MCR, technical support center, post accident sampling system, radiochemical laboratory, and hot counting room. The applicant also stated that all the instruments and switches installed in these equipment rooms are automatic or remote-operable from the MCR, thus access by operating staff is not required. DCD Table 12.3-3, "Projected Dose Rates for the Vital Areas at Various times after an Accident" (Sheets 2 and 3) provides radiation exposure as a cumulative mission dose for each task. The applicant committed to revise DCD Subsection 12.3.1.2.2 to show that, except for the MCR, the mission dose evaluations assume that workers use respiratory protection devices, thus only direct dose is considered. The staff confirmed that the applicant's analysis approach, described in the response to **RAI 429-3178, Question 12.03-12.04-27, (Parts 1 and 3)** are based on the guidance of Item II.B.2 of NUREG 0737 and are, therefore, acceptable. The DCD mark-ups provided by the applicant are based on the results of the analysis. As a result, the staff concludes that **RAI 262-1972, Question 12.03-12.04-16,** is resolved. **RAI 429-3178, Question 12.03-12.04-27 (Parts 1 and 3),** is identified as **Confirmatory Item 12.03-12.04-4** and the staff will confirm that this information is included in a future revision of the DCD.

During a review of US-APWR FSAR Tier 2, Section 3.11 "Environmental Qualification of Mechanical and Electrical Equipment," the staff identified that based on the equipment service times specified in Appendix 3D some equipment located in vital areas require access for repair, recalibration or replacement during post accident conditions and that these missions were not described in DCD Section 12.3. Therefore, the staff issued **RAI 262-1972, Question 12.03-12.04-16**, **(Part 2)** asking the applicant to provide additional information regarding these instruments. The applicant's response to **RAI 262-1972, Question 12.03-12.04-16, (Part 2)**, dated May 7, 2009, stated that only the main steam pressure transmitters were affected, and the location of these pressure transmitters was not subject to high dose rates. However, the staff review identified numerous examples of Environmentally Qualified (EQ) equipment with limited service life, yet expected to be in operation during a post LOCA environment, located in areas subject to potential large post accident dose rates. Therefore based on 10 CFR 50 GDC-19, and the regulatory guidance provided in NUREG-0800 and RG-8.8 C.1.12.3.5, **RAI 262-1972, Question 12.03-12.04-16**, **(Part 2)**, is considered closed, but the issues remain open. In follow up question **RAI 429-3178, Question 12.03-12.04-27-SQ2,** the staff has requested that the applicant show mission paths on the applicable plant layout drawings, and to describe the design exposure values associated maintaining qualified equipment. The applicant's response to **RAI 429-3178, Question 12.03-12.04-27,** (Part 2) dated September 28, 2009, stated that DCD Table 3D-2 "US-APWR Environmental Qualification Equipment List" was significantly revised as part of the response to **RAI 358-2462, Question 03.11-2,** dated July 10, 2009. This revision was primarily focused on providing the NRC additional detail regarding the location of equipment requiring qualification and additional details regarding the radiation dose to which the equipment must be qualified. The staff does not believe that the response provided in **RAI 358-2462, Question 03.11-2**, adequately addressed that question. The resolution of issues described in Section 3.11, impact the resolution of **RAI 429-3178, Question 12.03-12.04-27, (Part 2)**, therefore to ensure that the information in Section 12.3 accurately reflects the exposure needed to maintain EQ qualified instruments following an accident, pending resolutions of those issues, **RAI 429-3178, Question 12.03-12.04-27 (Part 2),** is associated with the above request, and the staff identified this as **Open Item 12.03-12.04-1.**

12.3.4.1.2 Radiation Shielding

The objective of the plant's radiation shielding is to minimize plant personnel and population exposures to radiation during normal operation (including AOOs and maintenance) and during accident conditions, while maintaining a program of controlled personnel access to and occupancy of radiation areas. The design also includes shielding, where necessary, to mitigate the possibility of radiation damage to materials. Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area shown in Figure 12.3-1.

The design of the primary shield located around the reactor vessel; 1) limits the radiation level from sources within the reactor vessel and the RCS, thus allowing limited access to the containment during normal and full-power operation; 2) minimizes neutron streaming to the containment free volume by incorporating a labyrinth style gap between the reactor vessel and the primary shield wall; 3) minimizes neutron activation of components and structural material; and 4) allows access for controlled access for inspections while limiting personnel exposure. Air cooling of the concrete is provided to prevent overheating, dehydration, and degradation of the shielding and structural properties of the primary shield.

The secondary shield surrounds the RCS equipment, including piping, pumps, pressurizer, and SGs to protect personnel from the direct gamma ray radiation resulting from activation and fission products in the reactor coolant fluid. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma ray radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full-power operation. Additional shielded compartments are provided for some components of the CVCS located inside containment.

The US-APWR DCD describes the use of labyrinth shields to limit personnel exposure to potentially high dose rates from fuel. COL Information Item 12.3(5) requires the COL applicant to discuss the administrative and access controls for these areas. Because the operational programs for administrative and access controls require plant-specific information that is beyond the scope of the requested DC, the staff finds inclusion of COL Information Item 12.3(5) acceptable.

Generic Safety Issue (GSI) 137, “Refueling Cavity Seal Failure,” and NRC Bulletin 84-03, “Refueling Cavity Water Seal,” called for reactor licensees to address the potential for inadvertent refueling cavity (RC) drain down via the cavity water seal as well as the associated potential for uncovering spent fuel, either stored or in transit. Other NRC and industry documents such as Information Notice No. 92-25, “Potential Weakness in Licensee Procedures for a Loss of the Refueling Cavity Water,” Information Notice 84-93, “Potential Loss of Water from the Refueling Cavity,” the Institute of Nuclear Power Operators Significant Operating Event Report 85-1 “Reactor Cavity Seal Failure,” and NUREG 1449, “Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States,” describe industry experience like valve positioning errors, and problems with freeze seals that could have resulted in significant events. Information Notice 87-13, “Potential for High Radiation Fields Following Loss of Water from Fuel Pool” stated that in the event of a rapid RC drain down event, high dose rates might result from loss of shielding from around irradiated core components, other than fuel, stored in the refueling cavity. US-APWR DCD Tier 2, Revision 2, Section 5.3.3, “Reactor Vessel Integrity” describes the use of a welded permanent RC seal to address GSI-137. However the applicant does not otherwise address how the US-APWR design features or required operational procedures preclude rapid loss of RC water inventory through fluid lines connected to the RC or RCS, such as the floor penetration provided for filling the RC described in DCD Tier 2, Revision 2, Section 9.1.4.2.2.2, “Reactor Refueling Operations.”

Since the number and location of irradiated fuel bundles that may be present in the RC was not specified, insufficient information was available for the staff to conclude that adequate shielding and cooling water would remain in the event of a rapid drain down of the RC to an elevation near the depth of irradiated fuel or irradiated internal vessel components. Therefore, in **RAI 262-1972, Question 12.03-12.04-17, (Part 6)**, the staff requested the applicant to describe the design features provide preclude an inadvertent rapid loss of water from the RC, from other than a seal leak, to provide the safe lay down location for a fuel bundle, an estimation of the dose rate to personnel in the area, and any COL information items required to prevent an event (i.e. restrictions on use of nozzle dams, or valves that need to be locked closed during fuel movement). The applicant’s response to **RAI 262-1972, Question 12.03-12.04-17, (Part 6)** dated May 7, 2009, asserted that because an inadvertent loss of excessive inventory from the RC is precluded by the use of administrative controls on interface valves and a pool level alarm which would initiate prompt operator action to maintain level, there was no need to specify a safe lay down location for fuel or to estimate dose rates from irradiated fuel or irradiated internal vessel components, so no changes to the DCD were needed.

US-APWR DCD Tier 2, Chapter 5, Chapter 9 and Chapter 16 contain insufficient information to allow the staff to identify administrative controls on valves or penetrations to the RC or RCS provided for preventing inadvertent rapid RC drain down. Also, since US-APWR DCD Tier 2, Chapter 5, Chapter 7 and Chapter 9 do not mention a RC pool level alarm, the applicant’s response implies a reliance on the SFP level alarm to indicate a change in RC level. However because TS 3.7.12 “Fuel Storage Pit Water Level” and TS 3.9.7 “Refueling Operations Water Level” do not require both the SFP Weir Gate and the fuel transfer tube gate valve to be open or for the SFP Level alarm to be operable while fuel is in the RC, the SFP level alarm may not be able to monitor the RC level under all conditions where fuel is out of the reactor vessel in the RC. In addition, insufficient information was available to allow the staff to ascertain that the facility design and operating procedures would ensure that sufficient water inventory and pumping capacity would be available for shielding and cooling of fuel in the RC, during a rapid RC drain down event.

Therefore, **RAI 262-1972, Question 12.03-12.04-17, (Part 6)** is considered closed, but the issues remain open, and in order to adequately address operating experience considerations as specified by 10 CFR 52.47(a)(22), in follow up question **RAI 524-4020, Question 12.03-12.04-35,** the applicant was asked to:

1. Identify any locations in the RC where more than one fuel bundle, including any in transit, may be out of the reactor vessel at the same time.

2. Provide information on each location in the RC, where a licensee could safely store spent fuel assemblies should inadvertent rapid RC drain down occur while one or more fuel bundles were out of the reactor vessel in the RC.

3. Provide the estimated dose rates and the methods used to determine those dose rates for each of the locations where water level at the lowest point following the drain down event is less than 10 feet above the fuel bundles that are out of the reactor vessel.

4. Justify the continued use of the Fuel Drop Accident Analysis methodology described in RG 1.183 Appendix B, if the volume of water in the safe storage area is less than the amount to needed to ensure complete coverage of the fuel bundles for the time allotted for ensuring containment closure.

5. Describe and provide the estimated dose rates, from any other irradiated components, that may be exposed during an inadvertent RC drain down event

The applicant's response to **RAI 524-4020, Question 12.03-12.04-35**, dated March 12, 2010 stated that there were two temporary fuel storage racks located on the RC wall that were capable of holding a total of six fuel bundles. They stated that based on the response to **RAI 507-3993, Question 09.01.04-16**, dated February 15, 2010, (which was evaluated under SRP Section 9.1.4) that the maximum leakage rate from the cavity drain valves was assumed to be one gpm, so any decrease in water level would be slow and could be restored by the refueling water recirculation pump, therefore they would have time to return any fuel in transit to the Reactor Vessel. Because assumptions, such as the leakage rates and makeup water capabilities used in the response to **RAI 507-3993, Question 09.01.04-16,** impact the resolution of **RAI 524-4020, Question 12.03-12.04-35**, the staff reviews of the responses to these questions have been coordinated. **RAI 524-4020, Question 12.03-12.04-35** is associated with the above request for SE Section 12.3, and the staff identified this as **Open Item 12.03-12.04-2.**

Because of high radioactivity levels from activation and contamination products, materials located in the SFPs, reactor vessel, and refueling cavities could create very high radiation areas if unshielded. Materials in the RC and SFP are normally covered with more than 10 feet of water and are inaccessible to personnel performing duties above the water surface. Radiation dose rates that pose a significant health risk could occur as a result of gaps in concrete around fuel transfer tubes, air entrainment in underwater tools, or movement of irradiated fuel or irradiated components near the isolation gates of the cask loading pit (CLP) or fuel inspection pit (FIP), if the pits were drained of water. DCD Tier 2, Revision 1, Section 12.3.2.2.8 discusses the gaps around the fuel transfer tube and the adjacent buildings, the resultant potential for high radiation fields and the shielding provided by the labyrinth structure in the access to the area around the transfer tube. However Figure 12.3-8, “Labyrinth for radiation protection around Fuel Transfer Tube” does not clearly depict the area bounded by the labyrinth. Also, DCD Tier 2, Revision 1, Figure 12.3-8, “Labyrinth for radiation protection around Fuel Transfer Tube” indicates that there is a valve operator assembly for the Fuel Transfer Tube Gate valve, running through the 52 foot penetration area but this figure does not show an offset to prevent direct streaming paths, and the text does not describe any design features to prevent streaming through hollow portions of the valve operator shaft. The NRC staff issued **RAI 262-1972, Question 12.03-12.04-17, (Part 2)** requesting the applicant to describe the shielding associated with this valve operator. The applicant’s response to **RAI 262-1972, Question 12.03-12.04-17,** (Part 2) dated May 7, 2009, stated that the valve operator assembly for the Fuel Transfer Tube gate valve is hollow and is filled with water up to the same level as the RC during fuel transfer, so no streaming pathways exist and that all submersible tools or equipment for handling irradiated material underwater are designed with penetrations to ensure filling with water for shielding. Confirmatory calculations performed by the NRC staff showed that dose rates around the fuel transfer tube valve reach rod penetration would be consistent with the stated zoning for the area. Because the information regarding design features to prevent a VHRA due to the fuel transfer gate valve operator conforms to the guidance in the SRP, the staff finds the response acceptable and, **RAI 262-1972, Question 12.03-12.04-17, (Part 2)** is closed.

The staff review identified that the Fuel Transfer Canal, the CLP and the FIP about the SFP and are separated from the SFP by weir gates. As discussed in RG 8.38, movement of fuel or irradiated components can cause very high dose rates when water shielding is not present. The NRC staff issued **RAI 262-1972, Question 12.03-12.04-17, (Part 3)** asking the applicant to describe the design features provided to prevent movement of irradiated components to areas connected to the SFP that may not be full of water. The applicant’s response to **RAI 262-1972, Question 12.03-12.04-17, (Part 3)** dated May 7, 2009, stated that management of the radiation exposure for personnel working near weir gates which are connected to drained pits will be controlled by the operational procedures (which is described in Section 12.5) and if an irradiated component is to be moved near a weir gate connected to the drained FIP or CLP, access into the pit or near the weir gate would be administratively prohibited. The guidance contained in RG 8.38 C.4.4 states that areas that could become VHRA during certain operational occurrences, such as dropped fuel, should be controlled to provide for ready evacuation of the area. ANSI/ANS HPSSC-6.8.1 states that detectors shall be located in areas subject to significant changes in dose rates, due to operational transients or maintenance activities and specifically mentions the cask handling areas in the Fuel Building. Operational Experience demonstrates that there have been a number of incidents where irradiated fuel or components caused unexpected changes in dose rate due to a failure to meet procedural fuel handling requirements. Therefore, **RAI 262-1972, Question 12.03-12.04-17, (Part 3)** is considered closed, but the issues remain open, and the staff issued follow up **RAI 429-3178, Question 12.03-12.04-28, (Part 2),** asking the applicant to provide additional area radiation monitors (ARM) in those areas susceptible to significant dose rate changes. The applicant’s response to **RAI 429-3178, Question 12.03-12.04-28, (Part 2)**, dated September 28, 2009, stated that DCD Subsections 12.3.4.1.2 and 12.3.4.1.8 describe the existence and give the functional description of the fixed SFP ARM, and DCD Subsection 12.3.4.1.2 includes provisions for an additional portable ARM in the cask handling area during work activities in that area. Because the information regarding design features to prevent an undetected VHRA near auxiliary fuel pits conforms to the guidance in the SRP, the staff finds the response acceptable and **RAI 429-3178, Question 12.03-12.04-28, (Part 2)** closed.

DCD Section 12.3.2.2.8 discusses the Spent Fuel Transfer Canal, and tube shielding design. This section notes that there are provisions for access and inspection. DCD Figure 12.3-8 “Labyrinth for radiation protection around Fuel Transfer Tube” does not clearly depict the area bounded by the shock absorber labyrinth, but the staff is unable to determine the location of physical barriers or removable shielding provided to limit personnel radiation exposure. Therefore the staff issued **RAI 262-1972, Question 12.03-12.04-17, (Part 1)**, requesting that the applicant clearly define the access points, physical barriers preventing access, any shielding that is other than permanent and radiation monitor locations used to control personnel exposure in this area. The applicant’s responses to **RAI 262-1972, Question 12.03-12.04-17, (Part 1)**, dated May 7, 2009, committed to adding the access points for accessing and inspecting the area around the fuel transfer tube to DCD Figure 12.3-1 “Radiation Zones for Normal Operation/Shutdown” Sheets 9 and 10. Since a high dose rate is expected to be present in the area around the fuel transfer tube during fuel transfer, the area is not usually accessed during that time and access to the area is tightly controlled by means of an entrance lock. However, because the area enclosed by the barriers depicted on DCD Figure 12.3-1, Sheets 9 and 10 is large and includes one quadrant of the electrical penetration area, DCD Figure 12.3-1 Sheets 9 and 10 contain insufficient detail to allow the staff to determine if there are any portions of the area accessible to a major portion of a person's body. Therefore, **RAI 262-1972, Question 12.03-12.04-17, (Part 1)** is considered closed, but the issues remain open, and the staff issued follow up question **RAI 429-3178, Question 12.03-12.04-28, (Part 1),** requesting the applicant to describe those penetrations into the barricaded areas indicated on Sheets 9 and 10, that are large enough to permit access to a major portion of a person's body, and what provisions are provided to thwart entry and those entries into the barricaded areas that will be required during the period of fuel movement. The applicant’s response to **RAI 429-3178, Question 12.03-12.04-28, (Part 1)**, dated September 28, 2009, stated that access to the barricaded areas on DCD Figure 12.3-1 Sheets 9 and 10 is controlled by gates and entry to these areas is allowed only through the issuance of a Radiation Work Permit. In addition, entry to these areas is prohibited during the period of fuel movement. Because the information regarding design features to prevent an undetected VHRA near the fuel transfer tube conforms to the guidance in the SRP, the staff finds the response acceptable and **RAI 429-3178, Question 12.03-12.04-28, (Part 1)** is closed.

The DCD Section 12.3.2.2.8 discusses the Spent Fuel Transfer Canal and tube shielding design. DCD Section 9.1.2 notes that there are no drains connected to the SFP. However, because neither chapter DCD Chapter 9, nor DCD Section 12.3.2.2.8 discuss the design features provided to prevent deliberate or inadvertent draining of the CLP or the FIP, while an irradiated fuel bundle is in the area, the staff issued **RAI 262-1972, Question 12.03-12.04-17, (Part 4)**, requesting the applicant to describe the design features provided to prevent inadvertent draining of the CLP and the FIP. The applicant’s response to **RAI 262-1972, Question 12.03-12.04-17, (Part 4)**, dated May 7, 2009, stated that the CLP and the FIP do not have floor drains and are drained by using a temporary pump, when needed. Because the information regarding design features to prevent an inadvertent draining of auxiliary fuel pits conforms to the guidance in the SRP, the staff finds the response acceptable and **RAI 262-1972, Question 12.03-12.04-17, (Part 4)** is closed.

DCD Tier 2, Revision 1, Section 12.2.1.2 “Sources for Shutdown” notes that in the reactor shutdown condition one of the significant sources requiring permanent shielding is the ICIS. The incore instrumentation consists of movable neutron detectors (MIDs) which are inserted into the core though the in-core instrumentation system nozzles located on the closure head. Figure

7.7-1 “Basic System for Insertion of Movable Neutron Detectors” indicates that at least some portion of the transit tube between the drive mechanism and the reactor head is not shielded. Industry experience shows that detectors can stick in the core, and require personnel to work at the location of the movable drive system to free the detectors. There have been personnel overexposure events due to problems with MIDS, therefore, the staff issued **RAI 262-1972, Question 12.03-12.04-18**, requesting the applicant to provide additional information regarding the personnel protective features provided for work on the ICIS. The applicant’s response to **RAI 262-1972, Question 12.03-12.04-18**, dated May 7, 2009, stated that if the ICIS detectors become stuck in the core, temporary shielding and portable ARMs will be used as necessary to protect personnel working near the MIDS drive unit. Because of the area between the reactor head and the ISIS drive area is located a considerable distance above the floor elevation and would be difficult to shield, **RAI 262-1972, Question 12.03-12.04-18**, is considered closed, but the issues remain open, and the staff issued follow up question **RAI 429-3178, Question 12.03-12.04-29,** requesting the applicant to describe any provisions for supporting temporary shielding or anchor points for portable barriers. The applicant’s response to supplemental question **RAI 429-3178, Question 12.03-12.04-29,** dated September 28, 2009, stated that the radiation protection program and the ALARA program would be used to limit personnel access and provide for exposure reducing measures, such as temporary shielding, as warranted when working on the MIDS. The content of the Radiation Protection and ALARA programs are provided by NEI 07-03A and NEI 07-08A as described in Section 12.5. The staff confirmed that the process described above conforms to the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA. Accordingly, the staff finds these features acceptable. Therefore, the staff concludes that **RAI 429-3178, Question 12.03-12.04-29,** is closed

Concrete thicknesses and radiation shielding capability for the safety-related structures are described in DCD Tier 1, Revision 2, Subsection 2.2.1 “Building Structures Design Description”, Table 2.2-2 and Figures 2.2-3 through 2.2-13, and DCD Tier 1, Revision 2, Subsection 2.7.4.1 “Liquid Waste Management System (LWMS)”, which notes that equipment used for storing and processing radioactive material are shielded in accordance with their design basis source term inventories. DCD Tier 1, Revision 2, Table 2.8-1, “Radiation Protection Inspections, Tests, Analyses, and Acceptance Criteria” describes the ITAAC or corresponding Design Acceptance Criteria for radiation protection.

DCD Tier 1, Revision 2, Subsection 2.7.6.4, “Light Load Handling System” notes that the refueling machine utilizes electrical interlocks to assure an adequate depth of water for shielding and DCD Tier 1, Revision 2, Table 2.7.6.4-2, “Light Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria” describes the ITAAC for assuring radiation shielding in the light load handling system.

In addition to the stated design specifications and ITAAC, DCD Tier 2, Revision 2, Section 14.2.12.2.4.15 “Biological Shield Survey Test” provides additional assurance of compliance with 10 CFR Part 20 by requiring that radiation surveys be performed in accessible areas outside the biological shield at less than 5 percent, 50 percent and 100 percent power levels to verify that radiation zones and associated occupancy times during power operation will be as defined in Chapter 12 of the DCD.

12.3.4.1.3 Ventilation

RG 8.8 contains guidance on ventilation design features acceptable to control airborne radioactivity levels and maintain personnel doses ALARA. The ventilation systems are designed to protect personnel and equipment from extreme environmental conditions, and to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA and within the applicable limits of 10 CFR Part 20. Further, the design ensures that the dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19. In containment, the sources of airborne radioactive material are primarily noble gases and fission products contained in the small volume of RCS leakage, as well as activation of Argon-40 into Argon-41. In areas outside of the CB, the source of airborne radioactivity for a room or area is primarily from equipment leakage within the specified areas. The design incorporates the following features to minimize this leakage and thereby reduce the sources of airborne radioactivity:

- Ventilation air is supplied directly to the clean areas of the plant and exhausted from the potentially contaminated areas, thereby creating a positive flow of air from clean areas to potentially contaminated areas.

- Negative or positive pressure is used appropriately in plant areas to prevent exfiltration or infiltration of possible airborne radioactive contamination, respectively.

- Equipment vents and drains are piped directly to a collection system, preventing contaminated fluid from flowing across the floor to a drain and creating a potential airborne contamination problem.

- Ventilating air is re-circulated only in areas outside the radiologically controlled area.

- Consideration is given to the possible disruption of normal airflow patterns by maintenance operations, and provisions are made in the design to prevent adverse airflow direction.

The requirements of 10 CFR Part 50, Appendix A (GDC 19) state that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 0.05 Sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident. The applicant included the MCR as a vital area. The control room dose analysis, as evaluated in Chapter 15 of this SE (accident analysis), demonstrates that the criteria of GDC 19 for limiting dose to control room operators to less than 0.05 Sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident, is met. Further, the MCR ventilation system is designed to provide controlled overpressure as well as recirculation and filtration of room air, helping to ensure that dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19.

Ventilation design features, such as the use of space coolers, maintenance of differential pressure gradients to prevent exfiltration of radioactive material, use of High Efficiency Particulate Activity (HEPA) filters and charcoal adsorption media where appropriate, are provided to protect personnel and equipment from extreme environmental conditions, and to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA. However, US-APWR DCD Section 12.3.3.2 regarding design criteria for ventilation systems does not describe the design features provided to control airborne particulate and iodine contamination resulting from evaporation of the RC water volume, or evaporation from exposed reactor components, during refueling activities. Therefore, in **RAI 262-1972, Question 12.03-12.04-20,** the staff requested the applicant to describe the design features for minimizing airborne contamination around the RC during refueling operations. The applicant's response to **RAI 262-1972, Question 12.03-12.04-20**, dated May 7, 2009, stated that since the features to maintain low airborne radioactivity in the containment are described in US-APWR DCD Tier 2, Subsection 9.4.6, "Containment Ventilation System", there was no need to describe the features in US-APWR DCD Tier 2, Chapter 12. Some Operating Experience available to the staff indicates that some current plants have experienced airborne contamination events in containment resulting from evaporation from the RC or drying of wetted RCS internal components exposed to air, that were poorly mitigated by the containment ventilation system. Because US-APWR DCD Subsection 9.4.6 only states that the capacity of the containment high volume purge system is sized to maintain acceptably low levels of radioactivity, including noble gases, during refueling operations, that it has filtration equipment on the exhaust and that the exhaust is equipped with a radiation monitor the staff considers **RAI 262-1972, Question 12.03-12.04-20**, closed, but the issue remains open, in **RAI 429-3178, Question 12.03-12.04-31,** the NRC staff asked the applicant to provide additional information regarding the design features provided around the RC to assure compliance with the requirements of 10 CFR 20. The applicant's response to **RAI 429-3178, Question 12.03-12.04-31**, dated September 28, 2009, stated that because experience demonstrates that the dose from airborne activity is normally not a significant contributor to the total doses, wearing a respiratory mask and installation of a temporary area exhaust equipment protect workers from airborne contamination caused by drying RCS internal components exposed to air. Because the applicant's response is consistent with the requirements of 10 CFR 20 Subpart H, and the guidance contained in RG 8.15, the staff finds the response acceptable. However, DCD Tier 2, Revision 2, Subsection 12.3 has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 429-3178, Question 12.03-12.04-31,** is identified as **Confirmatory Item 12.03-12.04-6** and the staff will confirm that this information is included in a future revision of the DCD.

12.3.4.1.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation and airborne radioactivity monitors are discussed in DCD Tier 2, Section 12.3 and Section 11.5, "Process Effluent Radiation Monitoring and Sampling Systems." The radiation monitoring system consists of:

- ARM System
- Airborne Radioactivity Monitoring System
- Process and Effluent Radiation Monitoring System
- Sampling System
- PAM System radiation monitors.

The process and effluent radiation monitoring system and sampling systems are described in DCD Tier 2, Section 11.5. The PAM variables are described in DCD Tier 2, Section 7.5, Revision 1, and DCD Tier 2 Revision 1, Table 7.5-3, "PAM Variables," which notes that in addition to portable air sampling equipment, portable survey instruments used for dose rate and radioactivity measurements should be Type E PAM equipment. Contrary to the information provided in DCD Tier 2, Table 7.5-3, Subsection 12.3.6 did not address the Type Classification requirements for portable dose rate and activity monitoring instrumentation. Therefore, in **RAI 262-1972, Question 12.03-12.04-13, Part 3**, the staff asked the applicant to describe the Type Classification of portable dose rate and airborne activity monitoring instruments. The applicant's response to **RAI 262-1972, Question 12.03-12.04-13, Part 3**, dated May 7, 2009, revised the DCD to indicate that the portable dose rate and activity monitoring instruments are Type E PAM. This change is consistent with the guidance contained in RG1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," therefore the staff concludes that **RAI 262-1972, Question 12.03-12.04-13, Part 3,** is resolved as closed.

The plant ARM equipment alerts operators and other station personnel to changing or abnormally high radiation conditions in the plant to prevent possible personnel overexposures and aid health physics personnel in keeping worker doses ALARA. The applicant stated that the installed ARMs supplement the personnel and area radiation survey provisions of the health physics program described in DCD Tier 2, Section 12.5 "Operational Radiation Protection Program". The ARMs assure compliance with the personnel radiation protection requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, and the guidance contained in RG 1.97, RG 8.2, Revision 0, "Guide for Administrative Practices in Radiation Monitoring", RG 8.8, RG 1.21 Revision 2 "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste", ANSI-N13.1-1999 and IEEE 497-2002 "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations".

In order to inform personnel of local dose rates in the area, ARMs include a local readout and audible alarm in addition to readouts and alarms in the MCR. The Waste Management System ARM alarm also alerts the personnel in the AB control room. The Containment High Range Monitors also indicate at the safety-related display console. Considerations for ARM locations include:

- Areas which are normally accessible, and where changes in plant conditions can cause significant increases in personnel exposure rate above that expected for the area.

- Areas which are normally and occasionally accessible where a significant increase in exposure rate resulting from operational transients or maintenance activities may occur.

- The containment area where the level of radioactivity needs to be monitored to detect the presence of fission products during a DBA.

SRP Section 12.03-12.04 states that ANSI/ANS HPSSC-6.8.1-1981 provides acceptable guidance on the location and design criteria of ARM systems including recommendations for the installation of monitors in the reactor, auxiliary, and radioactive waste buildings. DCD Tier 2, Revision 1, stated that the ARM system conformed to ANSI/ANS HPSSC-6.8.1. However, the applicant did not locate area monitors near the Mobile Liquid Radioactive Waste area, the

BATP, the charging pumps, or waste gas compressors, as recommended in ANSI/ANS HPSSC-6.8.1. Therefore, in **RAI 262-1972, Question 12.03-12.04-13, Part 1**, the staff asked the applicant to describe the installed shielding and ARMs for these areas. The applicant's response to **RAI 262-1972, Question 12.03-12.04-13, Part 1**, dated May 7, 2009, states that portable ARMs will be used prior to entering and performing work in the RHRS pump and heat exchanger areas, and that prior to entering the other areas which have elevated dose rates and require access, such as the Safety Injection (SI) pump area, portable ARMs will also be used, if necessary, and portable shielding would be used as needed. Because the applicant's response appeared to be inconsistent with the guidance provided the SRP and in ANSI/ANS HPSSC-6.8.1-1981, **RAI 262-1972, Question 12.03-12.04-13, Part 1,** is considered closed, but the issues remain open, and the staff issued follow up question **RAI 429-3178, Question 12.03-12.04-25, (Part 1)**, asking the applicant to provide additional information regarding the apparent deviation from ANSI/ANS HPSSC-6.8.1-1981 regarding the use of ARM in these areas. The applicant's response **RAI 429-3178, Question 12.03-12.04-25, (Part 1)**, dated September 28, 2009, stated that for areas with positive access control features, such as normally locked doors, or areas where a radiological hazard only exists during specific work activities, a fixed ARM is not required. Instead, a portable ARM is installed to warn occupants of a deteriorated radiological condition. Portable ARMs will be utilized in the following locations:

- Refueling platform
- Residual heat removal pump and heat exchanger areas
- Hot machine shop
- Heating, ventilation and air conditioning filter area
- Cask handling area
- Equipment decontamination area
- Safe shutdown panel area

The applicant committed to revising the DCD to include additional criteria, including specific areas, for the use of portable ARMs. Because the location and design criteria of these and other ARMs (including the use of local and remote monitor readouts and alarms) conforms to the criteria of ANSI/ANS Standard HPSSC-6.8.1-1981 and the guidance contained in RG 8.8, the staff finds the response acceptable, however, DCD Tier 2, Revision 2, Subsection 12.3.4.1.2 has not yet been updated to include the information provided in the response to the above questions. Therefore, **RAI 429-3178, Question 12.03-12.04-25, (Part 1),** is identified as **Confirmatory Item 12.03-12.04-1,** and the staff will confirm that this information is included in a future revision of the DCD.

ANSI/ANS HPSSC-6.8.1 Section 4.2.1 and Section 4.2.2 state that detectors are to be located in areas subject to significant changes in dose rates due to operational transients or maintenance activities, with Table 2 specifically mentioning the in core instrument area. Contrary to the guidance provided in SRP Section 12.3-12.4 and ANSI/ANS-HPSSC-6.8.1, the area between the reactor head and the ICIS drive area, which is unshielded, subject to significant dose rate transients, and accessible to personnel during containment entries is not provided with a radiation monitor to alert personnel to dose rates greater than those expected for the area during normal operation. Therefore, in **RAI 262-1972, Question 12.03-12.04-19,** (**Part 3)**, the staff asked the applicant to provide additional information regarding the use of ARM equipment in ICIS area. The applicant's response to **RAI 262-1972, Question 12.03-12.04-19,** dated March 3, 2009, stated that portable ARM equipment would be used to control personnel exposure in this area. However, the US-APWR DCD Tier 2, Table 1.9.2-12 "US-APWR Conformance with Standard Review Plan Chapter 12 Radiation Protection" does not state any exceptions to SRP Section 12.3-12.4 "Facility Design Features" Subsection 4 "Area Radiation and Airborne Radioactivity Monitoring Instrumentation," therefore, the staff considers that the issue addressed by the applicant's response to **RAI 262-1972, Question 12.03-12.04-19,** (**Part 3)** remains open, and in **RAI 429-3178, Question 12.03-12.04-30**, the NRC staff asked the applicant to provide additional information about the ARM for the ICIS area. The applicant's response to **RAI 429-3178, Question 12.03-12.04-30,** dated September 28, 2009, stated that Figure I2.3-1 "Radiation Zones for Normal Operation/Shutdown Site (Sheet 11 of 34)" would be revised to change the location of the ICIS ARM to the area between the Reactor Head and the concrete shield wall. The applicant's response is consistent with the guidance contained in the SRP. However, DCD Tier 2, Revision 2, Subsection 12.3, has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 429-3178, Question 12.03-12.04-30,** is identified as **Confirmatory Item 12.03-12.04-5** and the staff will confirm that this information is included in a future revision of the DCD.

The DCD did not discuss the calibration of the ARMs, other than to state that calibration would be performed in compliance with 10 CFR Part 20. Contrary to the guidance provided in RG-1.206 C.1.12.3.4, FSAR Section 12.3, does not provide any information regarding the calibration methods, calibration frequencies and their bases, therefore, in **RAI 262-1972, Question 12.03-12.04-13 Q4**, the staff asked the applicant to provide information on the calibration methods, frequency and their bases, of the installed area and airborne monitors. The applicant's response to **RAI 262-1972, Question 12.03-12.04-13, Part 4**, dated May 7, 2009, stated that the calibration methods and the associated calibration frequencies of the area and airborne monitoring instrumentation are deferred to the detailed design phase, and during the detailed design phase, the regulatory criteria used to establish the design will be incorporated into the procurement specifications and will be used to evaluate the vendor designs for the area and airborne radiation monitors. However, RG 1.206 Section C.1.12.4 specifically states that the applicant is to provide the calibration methods and frequencies for the installed ARM and Process Radiation Monitoring (PRM) equipment. Some of the monitors addressed in this question have automatic functions that initiate protective measures for the MCR, while others allow the MCR to direct manual actions that limit the dose impact to the public. Other monitors serve to alert plant operators of abnormal dose rates during the performance of their activities supporting AOO or accident mitigating actions. The applicant response that the Operational Radiation Protection Program will control the calibration of installed plant equipment is inappropriate because the Radiation Protection program controls portable instrumentation and equipment, but it is not applicable to installed ARMs and PRMs. Since no guidance has been provided to the COL applicant regarding the methodology to be used to determine ARM and PRM setpoints and calibrations, and DCD Chapter 12 does not contain a COL information item for establishing the ARM and PRM calibration procedures. Therefore, **RAI 262-1972, Question 12.03-12.04-13, Part 4,** is considered closed, but the issues remain open, and in follow up **RAI 429-3178, Question 12.03-12.04-25, Part 3** , the NRC staff requested that the applicant provide information on the calibration methodology and frequency for installed area radiation monitors.

The applicant's response to supplemental **RAI 429-3178, Question 12.03-12.04-25, Part 3**, dated September 28, 2009, stated that the DCD would be modified to state that alarm setpoints are controlled by plant procedures and the offsite dose calculation manual, where appropriate, and the methodology to determine the calibration interval and setpoints for the ARMs and Process and Effluent Radiation Monitors would be described in the DCD Tier 2, Section 7.2.2.7. The information provided by the applicant for describing the calibration methodology for radiation monitors conforms to the guidance in the SRP, however, the DCD has not yet been updated to include these changes. Therefore **RAI 429-3178, Question 12.03-12.04-25, Part 3** regarding the revision to DCD Section 12.3.4.1.9, is identified as **Confirmatory Item 12.03-12.04-2** and the staff will confirm that this information is included in a future revision of the DCD.

The requirements of 10 CFR 70.24 "Criticality accident requirements" specify the use of a monitoring system capable of detecting a criticality in designated areas where specified quantities of special nuclear material are handled, used, or stored. In lieu of installing a criticality monitoring system, the applicant has chosen to meet the design and analysis requirements specified in 10 CFR 50.68(b) to demonstrate the prevention of inadvertent criticality. Performance of a 10 CFR 50.68(b) analysis is an acceptable alternative to compliance with 10 CFR 70.24. Additional detail regarding the US-APWR 10 CFR 50.68(b) analysis is provided in DCD Tier 2, Revision 2, Section 9.1.1 "Criticality Safety of New and Spent Fuel Storage."

The requirements of 10 CFR 50.34(f)(2)(xvii) (corresponding to Item II.F.1(3) of NUREG-0737) specify, in part, that the control room must include instrumentation to measure, record, and read out containment radiation intensity (high level). Further guidance is provided in Item II.F.1(3) of NUREG-0737, which indicates that the reactor containment should be equipped with two physically separate radiation monitoring systems that are capable of measuring up to 1E+5 grays (Gy) per hour (1E+7 roentgen per hour) in the containment following an accident. In DCD Tier 2, Section 12.3.4.1.3 "General System Description," the applicant stated that the design incorporates four electrically independent ion chambers located inside the containment to measure high range gamma radiation. These detectors will be mounted on the inner containment wall in widely separated locations, and will have an unobstructed "view" of a representative volume of the containment atmosphere. The staff confirmed that the design and qualification of these monitors conforms to the guidance contained in RG 1.97; SRP Branch Technical Position (BTP) 7-10, "Guidance on Application of RG 1.97;" and NUREG-0737, Item II.F.1(3), with respect to detector range, response, redundancy, separation, onsite calibration, and environmental qualification. In addition, DCD Tier 1, ITAAC, Table 2.7.6.13-3, "Area Radiation and Airborne Radioactivity Monitoring Systems Inspections, Tests, Analyses, and Acceptance Criteria," provides assurance that the function, environmental qualification, seismic qualification, power supply and alarms for these safety related monitors will be inspected and verified prior to fuel load. Based on the above information provided in Tier 1 and Tier 2 of the DCD, the staff finds the monitors to be acceptable for demonstrating compliance with the requirements of 10 CFR 50.34(f)(2)(xvii).

The ARM equipment will be placed in selected areas and ventilation systems to give plant operating personnel continuous information about the airborne radioactivity levels throughout the plant. The airborne radioactivity monitors are located upstream of the filter trains to monitor representative radioactivity concentrations from the areas being sampled. The guidance in Section 12.3 of the SRP indicates that airborne radioactivity monitors should be able to detect the time integrated change of the most limiting particulate and iodine species equivalent to those concentrations specified in Appendix B of 10 CFR Part 20 (one DAC) in each monitored plant area within 10 hours (i.e., monitors should be sensitive enough to measure 10 DAC-hours). DCD Tier 2, Section 12.3.4.2.9 “Range and Alarm Setpoints,” states that airborne radioactivity monitoring system is capable of detecting 10 DAC-hours of particulate and iodine radioactivity from any compartment that has a possibility of containing airborne radioactivity and that normally may be occupied by personnel, taking into account dilution in the ventilation system. Because the applicant conforms to the guidance in SRP Section 12.3, the staff finds the above airborne monitor design description to be acceptable.

The guidance in SRP Section 12.3 addresses the criteria and methods to be employed by the applicant for obtaining representative in-plant airborne radioactivity concentrations in all work areas. The applicant has stated, in DCD Tier 2, Section 11.5.5 “Combined License Information” that the COL applicant will provide site-specific information on the extent to which the guidance provided by RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste” is followed. The airborne radiation monitors supplement the personnel and area radiation survey provisions of the plant health physics program and assure compliance with the personnel radiation protection requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70, and the guidance contained in RG 1.97, RG 8.2, RG 8.8, RG 1.21, ANSI-N13.1-1999 and IEEE 497-2002. The applicant has identified this as COL Information Item 11.5(1), which is evaluated in Chapter 11 of this SE.

Item III.D.3.3 of NUREG-0737 (corresponding 10 CFR 50.34(f)(2)(xxvii)) guidance states that each applicant should provide equipment and associated training and procedures for accurately determining the airborne iodine concentrations in areas within the facility where personnel may be present during an accident. The applicant has stated in DCD Tier 2, Section 12.3.4, that the COL applicant will address the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. The applicant has identified this issue as COL Information Item 12.3(1). Because the selection of portable instrumentation and training to address Item III.D.3.3 of NUREG-0737 are operational program elements, which are described in NEI 07-03A (discussed in Section 12.1 and Section 12.5 of this SE) and are beyond the scope of the requested DC, the staff finds inclusion of COL Information Item 12.3(1) acceptable.

12.3.4.2 Dose Assessment

The staff reviewed the applicant’s dose assessment contained in DCD Tier 2, Section 12.4, “Dose Assessment,” for completeness and conformance with the guidance contained in RG 1.206 and the criteria set forth in Section 12.03-12.04 of the SRP to verify that the applicant had either committed to follow the guidance of the applicable RGs and staff positions set forth in Section 12.3-4 of the SRP, or has provided acceptable alternatives. Where the DCD is consistent with the guidance in these RGs and staff positions, the staff can conclude that the relevant requirements of 10 CFR Part 20 have been met. In addition, the staff compared portions of the applicant’s dose assessment, for specific functions and activities, to the experience of operating PWR plants. The plant radiation protection program will ensure that radiation exposures to operating personnel shall not exceed the occupational dose limits specified in 10 CFR 20.1201.

In DCD Tier 2, Section 12.4, the applicant provided an assessment of the annual occupational radiation dose that would be received by the operating staff of a facility. DCD Tier 2, Tables 12.4.-1 through 12.4-8 provide estimated occupational doses for various jobs and inspections that would be performed in the plant during maintenance and refueling periods, as well as for power operations. DCD Tier 2, Section 12.4 does not contain a separate determination of doses attributable to airborne activity; however, NUREG-0713, “Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities”, Volume 30, Table 3.10 “Internal Dose (CEDE) Distribution 1994–2008” shows that experience at operating LWR demonstrates that the doses from airborne radioactivity are not a significant contribution to the total dose. The radiation protection program, as outlined in NEI 07-03A, which is discussed in Section 12.5, contains program elements for limiting internal personnel exposure which conforms to the guidance of RG 8.15 and the requirements of 10 CFR 20 Subpart H.

In performing the dose assessment, the applicant reviewed exposure data from similar operating plants in Europe, as well as U.S. operating plants to obtain a breakdown of the doses incurred within each dose assessment category referenced in RG 8.19, “Occupational Radiation Dose Assessment in Light-Water Reactor Plants - Design State Man-Rem Estimates,” Revision 1, dated June 1979. The applicant then adjusted these values to account for US-APWR design features. Based on its calculations, the applicant obtained an estimated annual dose of 0.7103 person-Sievert (71.03 person-rem).

The NRC compiles and publishes annual OREs from its licensees in NUREG-0713, “Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities”. Based on the information in NUREG-0713, Volume 27, which contains data compiled through 2005, and is the version referenced in DCD Tier 2, Section 12.4, the average annual exposure for U.S. PWR type plants is less than 1 person-Sievert (100 person-rem) per year. This average annual exposure information is consistent with the current version of NUREG-0713 Volume 30, which contains data compiled through 2008. The cumulative annual dose of 0.7103 person-Sievert (71.03 person-rem) for operating a US-APWR plant is consistent with the EPRI design guideline of 1.0 person-Sievert (100 person-rem) per year and compares favorably with current PWR experience. As discussed in NUREG-0713 Vol-27, average collective dose for U.S. PWRs was 0.79 person-Sievert (79 person-rem) and the median collective radiation exposure (CRE) for PWRs was 0.64 person-Sieverts (64 person-rem) in 2005. During this same period 25 percent of all PWRs had a CRE less than 0.44 person-Sieverts (44 person-Rem). NUREG-0713 Table 4.6 notes that the average Total Effective Dose Equivalent (EDEs) per Mega Watt-year (MW-y) for a number of plants similar to the US-APWR [1100 Mega Watt electric (MWe) 4 loop PWR], were significantly less than 0.1 person-Rem per MW-y. But, as stated in DCD Tier 2, Revision 2, Table 1.3-1, “Comparison of General Information and Reactor Core Characteristics”, the Gross electrical output Mega Watt electric (MWe) of the US-APWR is nominally 1,700 MWe while the current US PWR 4 loop plant is 1,186 MWe, which results in an estimated exposure of 0.044 person-rem per MW-y, for a 95 percent capacity factor, compared to the NUREG-0713 2005 average value of 0.09 person-rem per MW-y.

The conservative source term and shielding, the nuclear fuel performance, the facility layout, and the equipment and piping layout and design incorporated in the US-APWR design will contribute to lower plant exposures. Because the length of the fuel bundle has been increased from 12 feet to 14 feet, and the number of fuel bundles in the core has been increased from 193 to 257, the reactor thermal power increases from 3,411 MWt, to 4,451 MWt, while reducing the average linear heat rate kilo Watt per foot (kW/ft) from 5.4 kW/ft to 4.6 kW/ft, making the US-

APWR design less susceptible to axial offset anomaly, and the resultant excessive activated corrosion product concentrations during refueling outages.

The use of zinc injection in the US-APWR design will minimize the inclusion of corrosion and activation products into the corrosion film of piping and components exposed to RCS fluid, which will lead to lower dose rates throughout the plant. Elimination of high maintenance components contributes significantly to lower anticipated doses due to waste processing activities. The refueling process is labor intensive, so the incorporation of advanced technology reduces the dose associated with refueling. The US-APWR implements a long-term refueling cycle (about 24 months) which reduces the annual occupational radiation exposure by reducing the frequency of refueling operations. The design provisions supporting the use of the MHI remote ultrasonic testing machine for in service inspections (ISI) helps to reduce the exposure associated with ISI. The ORE resulting from unscheduled repairs on valves, pumps, and other components will be lower for the US-APWR than for current plant designs because of the reduced radiation fields and increased equipment reliability. Historically, special maintenance performed on SGs has resulted in significant personnel doses. The use of long service life alloy 690 in SG tubes will reduce the dose associated with SG ISI and special maintenance requirements.

SRP Section 12.3-12.4 specifies that the applicant, using the guidance contained in RG 8.19, is to provide a dose assessment, along with sufficient detail to explain the performance of the assessment process, used for evaluating dose-reducing changes in design and operations. DCD Tier 2, Revision 1, Tables 12.4-1 through 12.4-6, provide information regarding exposure estimates associated with routine activities during normal operation. However, based on the NRC staff industry experience, some of the assumptions and parameters regarding exposure estimates for routine plant surveillance activities described in these tables did not appear to be consistent with expected plant operations. Therefore the staff issued **RAI 262-1972, Question 12.03-12.04-14.** The applicant's response to **RAI 262-1972, Question 12.03-12.04-14**, dated May 7, 2009, rectified the inconsistent data, and committed to revising the DCD. The information provided in the DCD regarding Dose Assessment is consistent with the guidance provided in RG 8.19. The staff confirmed that Revision 2 to DCD Tier 2, Section 12.4 included this information. Accordingly, staff considers **RAI 262-1972, Question 12.03-12.04-14** closed.

12.3.4.3 Minimization of Contamination

The requirements in 10 CFR 20.1406 and the guidance in SRP Section 12.3-12.4 state that each licensee shall describe how they intend to minimize, to the extent practicable, contamination of the facility and of the environment, the generation of radioactive waste and how the design will facilitate decommissioning. As discussed in the preamble for the 10 CFR 20.1406 rule making published in Federal Register, Vol. 62, No. 139 (dated July 21, 1997) the intent of Section 20.1406 is to emphasize to a license applicant the importance, in an early stage of planning, for facilities to be designed and operated in a way that would minimize the amount of contamination, and that requirements are directed towards those making an application for a new license because it is more likely that consideration of design and operational aspects that would reduce dose and minimize waste can be cost-effective at that time. RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," contains a basis acceptable to the staff for complying with the requirements of 10 CFR 20.1406. Where the information provided by the applicant is consistent with this guidance, the staff can have reasonable assurance of compliance with 10 CFR 20.1406.

Because the DCD did not contain information that addressed the guidance provided in RG 4.21 about design features for minimizing contamination and radioactive waste generation, the staff issued **RAI 91-1496, Question 12.03-12.04-2**, asking the applicant to address compliance with 10 CFR 20.1406(b). As part of this question, the applicant was asked to include the methods of addressing the requirements of 10 CFR 20.1406 in the specific DCD sections that addressed the design features, as well as providing a summary in DCD Section 12.3 of these design features. The applicant's response to **RAI 91-1496, Question 12.03-12.04-2,** dated January 9, 2009, did include some information in DCD Tier 2, Revision 2, Chapter 12, but the DCD Tier 2, Revision 2, changes for Chapter 3, Chapter 5, Chapter 6 and Chapter 10 did not mention 10 CFR 20.1406, and the only reference to 10 CFR 20.1406 in Chapter 9 was to the SFP leakage detection system. Therefore it was not clear to the staff that the applicant has fully described the design provisions and program element requirements needed for compliance with 10 CFR 20.1406. **RAI No. 91-1496, Question 12.03-12.04-2,** is considered closed**,** but the issue it raised remained open**.** Consequently the staff issued follow-up **RAI 578-4483, Question12.03-12.04-37**, asking the applicant to fully describe in the specific DCD sections the design features provided to demonstrate compliance with 10 CFR 20.1406, and to summarize these features in DCD Section 12.3. The applicant's response to **RAI 578-4483, Question 12.03-12.04-37**, dated July 30, 2010, and with an amended response dated August 9, 2010, provided additional information about the design features provided to minimize contamination and reduce waste. The applicant committed to revising the DCD to include Table 12.3-8, "Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste", referencing NEI 08-08A "Guidance for Life Cycle Minimization of Contamination", adding DCD Section 12.3.1.3 "Minimization of Contamination and Radioactive Waste Generation". The staff has reviewed NEI 08-08A (ML093220530) and determined it to be acceptable. The commitment to add COL Information Item 12.3(10) requiring the COL applicant to address the site specific, operational and post-construction objectives and conceptual site model guidance to capture the operational and programmatic objectives guidance of RG 4.21. Because the operational programs and site specific conceptual model require plant-specific information that is beyond the scope of the requested DC, the staff finds inclusion of COL Information Item 12.3(10) acceptable. However, the proposed DCD mark up provided in the applicant's response did not include information related to a number of the items discussed in the response, and the design features provided to minimize contamination from some systems that contain radioactive material, were not clear. **RAI 578-4483, Question 12.03-12.04-37,** is associated with the above request and the staff identified this as **Open Item 12.03-12.04-3**.

As a result of recent industry experience with leakage from non-safety related buried piping, such as condensate transfer piping, the staff examined the DCD for design features specifically provided to address minimizing contamination from secondary side fluid system leakage, especially from buried piping. The staff noticed that the calculated secondary coolant activity levels presented in DCD Tier 2, Revision 2, Chapter 11, which were based on primary to secondary leakage rates much less than those allowed by TS, were greater than the qualitatively described activity levels in DCD Tier 2, Revision 2, Chapter 10. Therefore the staff issued **RAI 578-4483, Question 12.03-12.04-39**, asking the applicant to reconcile the calculated secondary coolant activity levels presented in DCD Tier 2, Revision 2, Chapter 11, with the description of secondary coolant activity levels presented in DCD Tier 2, Revision 2, Chapter 10. The applicant's response to **RAI 578-4483, Question 12.03-12.04-39**, dated July 30, 2010, committed to revising the DCD Chapter 10 sections describing the main steam and other secondary coolant system to include low level activity in the secondary coolant due to tritium and low volume primary to secondary leakage, as well as clarification in DCD Table 11.1-9 “Realistic Source Terms” about the assumed primary to secondary leakage rate. However, the DCD has not yet been updated to include the information provided in the response to the above question. Therefore, **RAI 578-4483, Question 12.03-12.04-39** is identified as **Confirmatory Item 12.03-12.04-10** and the staff will confirm that this information is included in a future revision of the DCD.

In addition, the staff also concluded that the applicant’s response to **RAI 91-1496, Question 12.03-12.04-2**, dated January 9, 2009, regarding contamination control design features for secondary coolant, was inconsistent with the secondary coolant activity levels calculated in DCD Tier 2, Chapter 11. Therefore the staff issued **RAI 578-4483, Question 12.03-12.04-38,** asking the applicant to fully describe the design features specifically provided to address minimizing contamination from secondary side fluid system leakage, especially from buried piping. The staff identified a number of secondary support systems, such as the Condensate Storage and Transfer System, the Auxiliary Steam System and the SG Blowdown System, as examples of systems that required further consideration by the applicant. The applicant’s response to **RAI 578-4483, Question 12.03-12.04-38**, dated July 30, 2010, which was amended August 9, 2010, provided additional information about the design features for some secondary side systems and components, but lacked sufficient detail to allow the staff to conclude that the question had been adequately addressed. **RAI 578-4483, Question 12.03-12.04-38** is associated with the above request and the staff identified this as **Open Item 12.03-12.04-4.**

In current operating reactors, the SFP has been the source of leaks that have resulted in extensive low level facility contamination. The US-APWR SFP has several design features that address this potential problem. For example, the SFP is located above the lowest elevation of the Fuel Storage Building and is equipped with a stainless steel liner. There are no SFP systems that are buried or routed through exterior boundaries. The fuel transfer tube between the Fuel Storage Building and the CB is capable of being inspected. The leakage detection system under the SFP provides coverage in case of a leak, and leak detection equipment in channels aid in identifying the location of the leak. Sumps that have the potential to collect spent fuel pool leakage are doubled-lined with non-porous material. In addition, walls and curbs are used around locations of potential leaks of contaminated fluids.

DCD Section 12.2.1 provided a brief description of the RWSAT, and the PMWT, which are located outside of the nuclear island block (AB, RB and CB). The RWSAT stores a portion of the water used to support refueling activities. The two PMWTs receive the distillate output of the BAE. In DCD Tier 2, Revision 1, Subsection 12.2.1.1.10, the applicant stated that these three tanks would be in concrete shielded enclosures. However, the applicant did not provide information regarding the configuration of the shielding and the resultant radiation zone near these tanks. **In RAI 144-1738, Question 12.02-12**, the staff asked the applicant to provide this information. The applicant’s response to **RAI 144-1738, Question 12.02-12**, dated February 6, 2009, (which was evaluated in Section 12.2) stated that they had changed the design to remove the concrete shielding surrounding these tanks. The applicant did not reflect the use of barriers, and the resultant radiation zone for the areas surrounding these tanks on DCD Tier 2 Figure 12.3-1 “Radiation Zones for Normal Operation/Shutdown Site (Sheet 1 of 34)”. **RAI 144-1738, Question 12.02-12**, was closed but the issue it raised remained open. In follow-up **RAI 427-2909, Question 12.02-21**, the staff asked the applicant for additional information regarding how these tanks would meet regulatory requirements and the applicable regulatory guidance. The applicant's response to **RAI 427-2909, Question 12.02-21**, dated September 28, 2009, (which was evaluated in Section 12.2) stated that the tanks would be located within a tank house, and the applicant committed to changing DCD Figure 12.3-1 to include the radiation zone information for the area around these tanks. But the response did not address the guidance of RG 4.21 related to 10 CFR 20.1406 "Minimization of Contamination," with respect to minimizing contamination of the environment. Therefore in a supplemental question, **RAI 532-4019, Question 12.02-29,** the staff asked the applicant to describe in DCD Tier 2, Section 12.3, the design features of the tank enclosure structure provided for compliance with 10 CFR 20.1406. The applicant's response to **RAI 532-4019, Question 12.02-29**, dated September 14, 2010, provided additional information about the design of the enclosure building, but it did not include a description of some of those provisions in the DCD and leakage detection and prevention measures for some below grade structures were not clearly defined. **RAI 532-4019, Question 12.02-29,** is associated with the above request, and the staff identified this as **Open Item 12.02-2.** This open item is also addressed in Section 12.2 above.

In addition to being a consideration in the design process, meeting the requirements of 10 CFR 20.1406 is also an operational concern. Programs and procedures should be in place to minimize contamination of the facility, minimize the contamination of the environment, and facilitate decommissioning. Because the DCD did not contain information that addressed the guidance provided in RG 4.21 about minimizing contamination and radioactive waste generation, the staff issued **RAI91-1496, Question 12.03-12.04-2**, asking the applicant to address compliance with the programmatic requirements of 10 CFR 20.1406(b). The applicant's response to **RAI 91-1496, Question 12.03-12.04-2**, dated January 9, 2009, included COL Information Item 12.1(6) requiring the COL applicant to perform periodic reviews of its operational practices for minimizing contamination, and COL Information Item 12.1(7) requiring the applicant to implement the tracking requirements of 10 CFR 50.75(g) and 10 CFR 70.25(g). Because it was not clear to the staff that the applicant had fully described the program element requirements needed for compliance with 10 CFR 20.1406, **RAI 91-1496, Question 12.03-12.04-2**, is considered closed, but the issue it raised remained open, and the staff issued follow-up **RAI 578-4483, Question 12.03-12.04-37**, asking the applicant to fully describe in the specific DCD sections the design features provided and required program elements needed to demonstrate compliance with 10 CFR 20.1406, and to summarize these features in DCD Section 12.3. The applicant's response to **RAI 578-4483, Question 12.03-12.04-37**, dated July 30, 2010, and with an amended response dated August 9, 2010, provided additional information about the design features provided to minimize contamination and reduce waste. The applicant committed to revising the DCD to include Table 12.3-8 "Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste", referencing NEI 08-08A "Guidance for Life Cycle Minimization of Contamination", and adding DCD Section 12.3.1.3 "Minimization of Contamination and Radioactive Waste Generation", as well as the commitment to add COL Information Item 12.3(10) requiring the COL applicant to address the site specific, operational and post-construction objectives and conceptual site model guidance of RG 4.21. The staff has reviewed NEI 08-08A and determined it to be acceptable (ML093220530). Because the operational programs and site specific conceptual model require plant-specific information that is beyond the scope of the requested DC, the staff finds inclusion of COL Information Item 12.3(10), COL Information Item 12.1(6) and COL Information Item 12.1(7) acceptable. However, the response provided by the applicant did not provide sufficient information to allow the staff to conclude that the guidance of RG 4.21 and the SRP has been adequately addressed. Therefore, **RAI 578-4483, Question 12.03-12.04-37,** has been identified as **Open Item 12.03-12.04-3**

12.3.5 Combined License Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the DCD:

Table 12-3
US-APWR Combined License Information Items

Item No.	Description	Section
12.3(1)	The COL applicant is responsible for the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.	12.3
12.3(2)	Deleted	12.3
12.3(3)	Deleted	12.3
12.3(4)	The COL applicant is to provide the site radiation zones that are shown on the site-specific plant arrangement plan.	12.3
12.3(5)	The COL applicant is to discuss the administrative control of the fuel transfer tube inspection and the access control of the area near the seismic gap below the fuel transfer tube.	12.3

Table 12-4
US-APWR Combined License Information Items

Item No.	Description	Section
12.4(1)	For multiunit plants, the COL applicant is to provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s).	12.4

Combined License Information Items not identified in Table 1.8-2 of the DCD:

Item No.	Description	Section
12.3(6)	If the COL applicant adopts the Mobile Liquid Waste Processing System, the COL applicant is to provide information about the radiation protection aspects of the system and to indicate how the system is consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.1.12.3.2 and RG 1.69	12.3
12.3(7)	If the COL applicant adopts the Mobile Liquid Waste Processing System, the COL applicant is to provide information about prevention and detection of contamination of the environment and minimization of decommissioning costs and to explain how the system meets the requirements of 10 CFR 20.1406 and RG 4.21.	12.3
12.3(8)	If the COL applicant adopts the Mobile Liquid Waste Processing System, the COL applicant is to confirm the radiation zone(s) where the system is installed in and to revise Figure 12.3-1, if necessary.	12.3
12.3(9)	In order to ensure that the B.A. evaporator room does not become a VHRA during the end of cycle, the COL Applicant is to stipulate a need for routine surveillance in the Radiation Protection Program. In the event that the routine surveillance shows an increase in dose level, the COL Applicant must provide an appropriate strategy to sufficiently reduce the dose rate below the criteria for a VHRA.	12.3
12.3(10)	The COL Applicant will address the site-specific design features, operational, post-construction objectives, and conceptual site model guidance of Regulatory Guide 4.21.	12.3

The identified COL license information items have been reviewed by the staff and have been found to be relevant, complete, appropriate for this section and focused on matters that may be a significant issue in any COL application referencing the DCD, and therefore, conform to the guidance contained in the SRP.

12.3.5 Conclusions

For the reasons set forth above, and with the exception of the open and confirmatory items discussed above and listed below, the applicant’s radiation protection design features help maintain occupational radiation exposures within regulatory limits and ALARA, comply with the requirements of 10 CFR 20.1101(b); the definition of ALARA in 10 CFR 20.1003; the dose limits of 10 CFR 20.1201, 10 CFR 20.1202, 10 CFR 20.1203, and 10 CFR 20.1204; and the non-effluent limits in 10 CFR 20.1301 and 10 CFR 20.1302; and conform to the guidance provided in RG 8.8 and 8.10. In addition, with the exception of the open and confirmatory items discussed above and listed below, the design features comply with the radiation exposure and radiations source control requirements in 10 CFR 20.1406, 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1801, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1905 “Exemptions to labeling requirements." Many of these design features have been incorporated as a result of the applicant's radiation design review and from radiation exposure experience gained during the operation of other nuclear power plants. The staff confirmed that these design features are consistent with those contained in RG 8.8 and RG 8.38 and are, with the exception of the open and confirmatory items discussed above and listed below, acceptable.

The plant design and layout facilitates the control of access to and work within plant areas in accordance with the requirements of 10 CFR 50.34(f)(2)(vii), 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, and 10 CFR 20.1903 and access control alternatives in the Standard TS - Westinghouse Plants (NUREG-1431, Revision 3). Except for the matters identified in **Confirmatory Item 12.03-12.04-2**, and **Open Item 12.03-12.04-4**, regarding mission dose estimates for some EQ equipment, the staff finds the plant design and layout acceptable.

The general shield design methodology and source term inventories used by the applicant are similar to those of operating reactors. The basic radiation transport analysis used for the applicant's shield design is based on approved analysis codes, such as Microshield 7, RSICC Computer Code Collection CCC-710, MCNP5: Monte Carlo N-Particle Transport Code System, RSICC Computer Code Collection CCC-564, GGG-GP: Kernel Integration Code System Multigroup Gamma-Ray Scattering Using the GP Buildup Factor and RSICC Computer Code Collection CCC-650, DOORS3.2: One, Two, and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System. All concrete shielding in the plant will be constructed in general compliance with RG 1.69. As discussed above, based on the information provided by the applicant regarding potential dose rates from irradiated fuel and irradiated internal vessel components due to a rapid RC drain down **RAI 524-4020, Question 12.03-12.04-35,** is associated with the above issue, and the staff identified this as **Open Item 12.03-12.04-2.** Therefore the staff is unable to confirm at this time that the overall design approach, as well as the specific examples of design features, demonstrate compliance with 10 CFR 50 GDC 61, and the guidance in RG 8.38 and RG 8.8.

The ventilation system is designed to ensure that plant personnel are not inadvertently exposed to airborne contaminants in excess of the limits provided in 10 CFR Part 20. The applicant intends to maintain personnel exposures ALARA by (1) maintaining airflow from areas of potentially low airborne contamination to areas of higher potential concentrations, (2) ensuring negative or positive pressures to prevent exfiltration or infiltration, respectively, of potential contaminants, and (3) conforming to all other applicable guidance contained in RG 8.8. Therefore, with the exception of the open and confirmatory items discussed above and listed below, the applicant's ventilation design features for radiation protection help maintain OREs within regulatory limits and ALARA, ensure that that the spread of airborne contamination is minimized or contained, are consistent with the guidance contained in RG 1.52 and RG 8.8 and, accordingly, demonstrate compliance with the requirements of 10 CFR Part 20.

The applicant's area radiation monitoring system is designed so that it will (1) monitor the radiation levels in areas where radiation levels could become significant and where personnel could be present, (2) alarm when the radiation levels exceed preset levels to warn of increased radiation levels, and (3) display data using the process information and control system. To meet these objectives, the applicant plans to use 12 permanent area monitors located in areas where personnel may be present and where radiation levels could become significant. The installed ARM system is supplemented by seven specifically identified locations where portable ARM equipment is required to be used during work activities. The design objectives of the airborne radioactivity monitoring system are (1) to assist in maintaining occupational exposure to airborne contaminants ALARA, (2) to check on the integrity of systems containing radioactivity, and (3) to warn of unexpected release of airborne radioactivity. The applicant will install airborne radioactivity monitors in areas of the plant where there is a potential for airborne radioactivity. These airborne radioactivity monitors will have the capability to detect DAC of the most restrictive particulate and iodine radionuclides in the area or cubicle of lowest ventilation flow rate within 10 hours.

Except for the matters identified in **Confirmatory Item 12.03-12.04-3**, and **Confirmatory Item 12.03-12.04-4** the staff finds that the objectives and location criteria of the area and airborne radiation monitoring systems is in conformance with those portions of 10 CFR 20.1501 "General," 10 CFR 50.34, "Contents of applications; technical information," 10 CFR 50.68, "Criticality accident requirements," as well as RG 1.97, and RG 8.8, related to radiation and airborne radioactivity monitoring.

The objective of the applicant's accident radiation monitoring system is to provide the capability to assess the radiation hazard in areas that may be occupied during the course of an accident. The installed accident radiation monitors will have emergency power supplies. The systems will be designed for use in the event of an accident in terms of location, usable instrument range and the environment the instrument can withstand, and, accordingly, meet the requirements of 10 CFR 50.34(f)(2)(xvii), Item II.F.1(3) of NUREG-0737, RG 1.97, and BTP 7-10. On the basis of its review of the information on radiation protection design (including facility design features, shielding, ventilation, and area radiation and airborne radioactivity monitoring instrumentation) supplied by the applicant for the US-APWR, as described above, the staff concludes that the applicant has committed to follow the guidance contained in the RGs and staff positions set forth in Section 12.03-12.04 of the SRP. Because the DCD is consistent with the guidance provided in these RGs and staff positions, with the exception of the open and confirmatory items discussed above and listed below, the staff concludes that the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70 have been met.

The staff further finds that the dose assessment for the US-APWR conforms to the guidance contained in RG 1.206 and in Section 12.03-12.04 of the SRP. This dose assessment also conforms to the intent of the guidance in RG 8.19. While the applicant did not conform to all the details in RG 8.19, they provided information on all dose significant activities that occur during normal operations and refueling. By addressing the anticipated occupational radiation exposures due to normal and anticipated inspection and maintenance, and by incorporating design features to reduce occupational radiation exposures, the applicant has shown that the US-APWR is designed to operate within the occupational dose limits specified in 10 CFR 20.1201. Because exposure to construction workers from operating units requires plant-specific information that is beyond the scope of the requested DC, the staff finds inclusion of COL Information Item 12.4(1) acceptable. Accordingly, the staff finds the material contained in DCD Tier 2, Section 12.4 acceptable with respect to dose assessment.

Based on the information provided by the applicant, at this time, the staff is unable to determine that the design features provided conform to the guidance contained in RG 4.21, RG 1.206, and in Section 12.03-12.04 of the SRP for implementing the minimization of contamination design philosophy. The staff is unable to confirm at this time, that the overall design approach, as well as the specific examples of design features (for such systems as the MLWPS, the RWSAT and

PMWT, the secondary coolant systems, and other auxiliary systems) demonstrate compliance with the guidance contained in RG 4.21 and the requirements of 10 CFR 20.1406. Accordingly, the staff has identified **Open Item 12.03-12.04-2, Open Item 12.03-12.04-3, and Open Item 12.03-12.04-4** related to compliance with 10 CFR 20.1406.

With the exception of the open and confirmatory items discussed above and listed below, on the basis of the information provided in the US-APWR, DCD Revision 2, on radiation protection design, occupational dose assessment, and minimization of contamination, as described above, the staff concludes that the applicant has committed to follow the guidance contained in the RGs and staff positions set forth in Section 12.03-12.04 of the SRP. Because the DCD is consistent with the guidance in these RGs and staff positions, the staff concludes that the relevant requirements of 10 CFR Part 20, 10 CFR Part 50 and 10 CFR Part 70 have been met, except for the matters identified in Confirmatory Items:

RAI 429-3178 Question 12.03-12.04-25 Part 1	**Confirmatory Item 12.03-12.04-1**
RAI 429-3178 Question 12.03-12.04-25 Part 3	**Confirmatory Item 12.03-12.04-2**
RAI 429-3178 Question 12.03-12.04-26	**Confirmatory Item 12.03-12.04-3**
RAI 429-3178 Question 12.03-12.04-27 Part 1, 3	**Confirmatory Item 12.03-12.04-4**
RAI 429-3178 Question 12.03-12.04-30	**Confirmatory Item 12.03-12.04-5**
RAI 429-3178 Question 12.03-12.04-31	**Confirmatory Item 12.03-12.04-6**
RAI 524-4020 Question 12.03-12.04-33	**Confirmatory Item 12.03-12.04-7**
RAI 524-4020 Question 12.03-12.04-34	**Confirmatory Item 12.03-12.04-8**
RAI 524-4020 Question 12.03-12.04-36	**Confirmatory Item 12.03-12.04-9**
RAI 578-4483 Question 12.03-12.04-39	**Confirmatory Item 12.03-12.04-10**

and **Open Items**:

RAI 429-3178 Question 12.03-12.04-27, Part 2	**Open Item 12.03-12.04-1**
RAI 524-4020 Question 12.03-12.04-35	**Open Item 12.03-12.04-2**
RAI 578-4483 Question 12.03-12.04-37	**Open Item 12.03-12.04-3**
RAI 578-4483 Question 12.03-12.04-38	**Open Item 12.03-12.04-4**

SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Application Without Inspections, Tests, Analyses, and Acceptance Criteria" states that in the absence of ITAAC, "fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability at the COL stage. The DCD specifies that the COL applicant will have a radiation protection program that meets the requirements of NEI 07-03A, which provides one acceptable method of describing the radiation protection program. The staff has reviewed NEI 07 03A and determined it to be acceptable (ML091490684). The applicant has committed to following the guidance of RG 4.21 to minimize contamination. Portions of RG 4.21 and NEI 08-08A address implementation of the program requirements of 10 CFR 20.1406. The staff has reviewed NEI 08-08A and determined it to be acceptable (ML093220530). Because the operational programs and site specific conceptual model require plant-specific information that is beyond the scope of the requested DC, the staff finds inclusion of COL Information Item 12.3(10), 12.1(6) and 12.1(7) acceptable.

12.5 Operational Radiation Protection Program

12.5.1 Introduction

The operational radiation protection program for a nuclear power facility assures that exposures of plant personnel to radiation are controlled and minimized. The administration of the radiation protection program and the qualifications of the personnel responsible for conducting various aspects of the radiation protection program and for handling and monitoring of radioactive material are important components of the program. Adequate equipment, instrumentation and facilities must also be provided for performing (1) radiation and contamination surveys, (2) in-plant airborne radioactivity monitoring and sampling, (3) area radiation monitoring, and (4) personnel monitoring. Procedures and methods of operation, including those for ensuring that ORE will be ALARA, must be in place. This includes procedures used in normal operation, refueling, ISI, handling of radioactive material, spent fuel handling, routine maintenance, and sampling and calibration related to radiation safety.

12.5.2 Summary of Application

DCD Tier 1: There are no DCD Tier 1 entries for this area of review.

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 12.5, summarized here in part, as follows:

The applicant states in Tier 2 Section 12.5 that the subject of this section will be addressed by the COL applicant.

ITAAC: There are no ITAAC for this area of review.

TS: There are no TS for this area of review.

COL information or action items - (See Subsection 12.5.5 below).

Technical Report(s): There are no technical reports associated with this area of review.

Topical Report(s): There are no topical reports associated with this area of review.

US-APWR Interface Issues identified in the DCD: There are no US-APWR interface issues associated with this area of review.

Site Interface Requirements Identified in the DCD: There are no site interface requirements associated with this area of review.

Cross-cutting Requirements (Three Mile Island [TMI], Unresolved Safety Issue [USI]/Generic Safety Issue [GSI], Op Ex): There are no cross-cutting issues associated with this area of review.

12.5.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 12.5 of NUREG-0800, the SRP.

12.5.4 Technical Evaluation

The US-APWR DCD Tier 2 Revision 2 Section 12.1 states that the COL applicant is to provide the radiation protection operational program, as described in NEI 07-03A. Administrative procedures and practices related to maintaining exposures ALARA are to be employed using NEI 07-08A "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)." The staff has reviewed NEI 07-08A (ML093220178), and determined it to be acceptable. The US-APWR DCD Tier 2, Section 12.5 states that the COL applicant is to provide the operational radiation protection program for ensuring that occupational radiation exposures are ALARA. It states that the program consists of the following:

- A detailed management policy.
- An organizational structure with clearly defined responsibilities.
- Definition and description of all facilities, including laboratories and office spaces.
- Definition and description of the monitoring instrumentation and equipment.
- Definition and description of the personnel protective clothing and equipment, including the necessary inventory of supplies.
- Definition and description of other protective equipment, such as portable ventilation systems, temporary shielding, etc..
- Procedures on radiological surveillance.
- Procedures on methods to maintain exposures ALARA.
- Procedures on posting and labeling.
- Procedures on access control.
- Procedures on radiation work permits.
- Procedures on personnel monitoring.
- Procedures on dose control.
- Procedures on contamination control.
- Procedures on respiratory protection.
- Procedures on radioactive material control.

- Procedures on radiation protection training.
- Quality assurance programs in effect

12.5.5 Combined License Information Items

There are none listed in DCD Tier 2, Revision 2, Section 12.5; instead the applicable items are listed under DCD Tier 2, Revision 2, Section 12.1.

Table 12-5
US-APWR Combined License Information Items

Item No.	Description	Section
	None stated	12.5

COL information items not identified in Table 1.8-2 of the DCD: None

12.5.6 Conclusions

SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Application Without Inspections, Tests, Analyses, and Acceptance Criteria" states that in the absence of ITAAC, "fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability at the COL stage. The DCD specifies that the COL applicant will have a radiation protection program that meets the requirements of NEI 07-03A, which provides one acceptable method of describing the radiation protection program. The staff has reviewed NEI 07-03A (ML091490684) and determined it to be acceptable. NEI 07-08A "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)" "Guidance for Life Cycle Minimization of Contamination" as a reference. The staff has reviewed NEI 07-08A (ML093220178), and determined it to be acceptable. Because the Radiation Protection and ALARA programs require plant specific information that is beyond the scope of the DCD the staff finds that the identification of the required elements of those programs, in conjunction with use of NEI 07-03A and NEI 07-08A, describes the required programs in a manner that conforms to the guidance in the SRP.